MULTI-RECYCLING OF TRANSURANIC ELEMENTS IN A MODIFIED PWR FUEL ASSEMBLY

A Thesis

by

ALEX CARL CHAMBERS

Submitted to the Office of Graduate Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

August 2011

Major Subject: Nuclear Engineering

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Approved by:

Chair of Committee,	Jean C. Ragusa
Committee Members,	Pavel V. Tsvetkov
	Guido Kanschat
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ABSTRACT

Multi-Recycling of Transuranic Elements in a Modified PWR Fuel Assembly.

(August 2011)

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Chair of Advisory Committee: Dr. Jean C. Ragusa

The nuclear waste currently generated in the United States is stored in spent fuel pools and dry casks throughout the country awaiting a permanent disposal solution. One efficient solution would be to remove the actinides from the waste and transmute these isotopes in a fast spectrum reactor. Currently this technology is unavailable on a commercial scale and a considerable amount of research and development is still required. An alternate solution is to reprocess and recycle the used fuel in thermal reactors, creating new fuel while reducing the amount of waste and its impact to the environment. This thesis examines the possibility of multi-recycling the transuranics (Pu, Np, Am, and Cm) in a standard pressurized water reactor (PWR). Two types of recycling strategies will be examined: one where Pu, Np, and Am are recycled (TRU-Cm) and a second where the previous isotopes as well as Cm are recycled (TRU+Cm). To offset the hardened neutron spectrum that results from the inclusion of the transuranics, a smaller fuel pin is employed to provide additional moderation.

Computer simulations are used to model the in-reactor physics and long-term isotopic decay. Each fuel type is assessed based on the required ^{235}U enrichment, void coefficient, transuranic production/destruction, and radiotoxicity reduction as compared to a UOX and MOX assembly.

It is found that the most beneficial recycling strategy is the one where all of the transuranics are recycled. The inclusion of Cm reduces the required ^{235}U enrichment, compared to the other multi-recycled fuel and, after a significant number of recycles,

can result in the required enrichment to decrease. This fuel type also maintains a negative void coefficient for each recycle. The void coefficient of the fuel type without Cm becomes positive after the third cycle. The transmutation destruction of the two multi-recycled assemblies is less than that of a MOX assembly, but the transmutation efficiency of the multi-recycled assemblies exceeds the MOX assemblies. The radiotoxicity of both multi-recycled assemblies is significantly lower than the UOX and MOX with the TRU+Cm fuel being the lowest. When Curium is recycled only 28,000 years are required for the radiotoxicity of the waste to reach that of natural Uranium and when Cm is not recycled, the amount of time increases to 57,000 years.

To my family in gratitude for their unending support and love

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CHAPTER I

INTRODUCTION

In the United States, the waste from commercial nuclear reactors is currently stored in dry storage casks or in the spent fuel pool on sites throughout the the country. This fuel has been irradiated in a reactor core and is being stored awaiting a final permanent solution such as storage at a geological repository or reprocessing for further use. The United States has recently decided to re-examine the proposed geological repository Yucca Mountain, and to consider alternative possibilities. In the meantime the spent fuel will remain in dry casks or spent fuel pools and the postponement of a more permanent solution continues to be an hindrance for the expansion of nuclear energy.

An alternative to storing the used nuclear fuel in a geological repository is to reprocess and reuse the spent fuel. Currently, in some European countries, Plutonium (Pu) from spent fuel is extracted and used to create Mixed Oxide (MOX) fuel. The United States is examining the possibility of using MOX fuel with some MOX assemblies being tested at the Catawba Nuclear Station in South Carolina. After one recycle the Pu as well as any additional minor actinides and fission products generated in the MOX fuel are then discharged as waste. Recycling Pu does reduce the amount of waste sent to the repository slightly but does not address the need to reduce the amount of minor actinides that must be stored. These minor actinides have an appreciable contribution to the long term radiotoxicity of the spent fuel. If the minor actinides as well as the Pu from used UOX fuel could be removed from the waste stream, the quantity and radiotoxicity of waste could be further reduced.

The journal model is *Nuclear Science and Engineering*.

The purpose Advanced Fuel Cycle Initiative (AFCI), proposed by the Department of Energy (DOE), is to examine solutions for the spent fuel issue. The objectives of this program are:[1]

- Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste material;
- Enhance the proliferation resistance of the fuel cycle with improved spent fuel management technologies;
- Enhance energy security by extracting energy recoverable in spent fuel and depleted Uranium ensuring Uranium does not become a limiting resource for nuclear energy;
- Improve fuel cycle management, while continuing competivite fuel cycle economices and excellent safety performance of the entire nuclear fuel cycle.

The work in this Thesis falls under the first objective, specifically, reducing the radiotoxicity of spent nuclear fuel by removing and transmuting the transuranic elements (Pu, Np, Am, and Cm). The possibility of multi-recycling the transuranics in a pressurized water reactor (PWR) is examined. Two types of recycling scenarios have been pursued, one in which Plutonium, Neptunium, and Americium are recycled and a second one where the previous elements along with Curium are recycled. The fuel assemblies developed in this Thesis are compatible with current PWR cores, with the only modification being the diameter of the fuel pin. Each fuel type will be recycled multiple times to attempt to achieve an equilibrium in the fuel composition so that these assemblies can be used to manage the minor actinides inventory indefinitely.

A successful design is considered to be one in which the enrichment of ^{235}U remains below 5 w/o, has a negative void coefficient during a LOCA, and is beneficial

in reducing the radiotoxicity of the spent fuel. To compare the recycled fuel assemblies to that of a standard Uranium oxide (UOX) the concept of energy equivalence has been employed. The ingestion toxicity of both recycled fuels are compared to that of a UOX fuel and a mixed oxide fuel (MOX). The transmutation efficiency of both recycled fuel types is also compared. Transmutation efficiency is defined as the amount of transuranics destroyed relative to the initial amount charged at each cycle.

CHAPTER II

LITERATURE SURVEY

Previous work in the area of partitioning and transmutation of transuranics (Pu, Np, Am, Cm) serves as the foundation and provided direction for the work carried out in this Thesis. Work that also examined on the effects of reducing the diameter of a fuel pin in terms of performance of a UOX and MOX fuel pin was also reviewed. A non-exhaustive summary is included in the sections that follow.

A. Partitioning and Transmutation of Transuranics

Countries throughout the world have examined the benefits of partitioning and transmuting (P&T) Plutonium as well as the other minor actinides that are created in commercial nuclear power plants.[2–4] Removing the transuranics from the waste for use in new nuclear fuel utilizes the available fissile material as well as reduces the amount and toxicity of the waste. Three methods are primarily examined to be employed in P&T of the transuranics are; Light Water Reactors (LWR), Fast Reactors, and Accelerator Driven System (ADS). Although promising, fast reactors and ADS are currently unavailable on a commercial scale and still require a considerable amount of research and development.[5] In the meantime, LWRs could still be used to transmute the transuranics elements. As can be seen in Table II-I, the largest component of the transuranics in the waste is Plutonium which is currently recycled in MOX fuel in France and Japan amongst others. In these countries, the Plutonium is only recycled once before it is discharged awaiting a final solution. This single recycle pass provides only limited benefits in terms of radiotoxicity.[6]

Table II-I. Representative weight percent (wt%) of each transuranic element discharged from a UOX fuel assembly (standard UOX 3.86 wt% ^{235}U burnup of 45 GWd/tHM).

Element	Weight Percent%
Pu	87.1%
Np	6.5%
Am	5.5%
Cm	0.9%

1. Pu-Only Multi-Recycling

Because of the limited benefits derived from recycling Plutonium only one time, studies have examined the possibility and benefits of multi-recycling Plutonium.[7,8] It was found that it is possible to recycle Plutonium multiple time in a standard pressurized water reactor (PWR). The main drawback to multi-recycling Pu is the buildup of the non-fissile, even Plutonium isotopes with each recycle. Because of this the enrichment of ^{235}U or the amount of Pu in the fuel must be increased. There is a legal limit on the enrichment of ^{235}U in commercial fuel at 5 w/o and increasing the Pu content beyond 8 to 12 w/o, depending on the design and isotopics of the fuel, results in a less negative and eventually positive void coefficient. LWRs cannot be licensed to operate with a positive void coefficient, thus limiting the amount of Pu allowed in MOX fuel. It was found that Pu could be multi-recycled in a homogeneous assembly (all of the fuel rods in the assembly are of the same composition), stabilizing the Pu inventory while keeping the Pu and ^{235}U enrichments below the set limits.[7]

Plutonium can also be recycled in a heterogeneous assembly, where the outer fuel pins contain Plutonium in an inert matrix (MgO-ZrO_2) and the inner pins contain

UOX. Inert matrix fuel (IMF) pins avoid breeding additional Plutonium due to the absence of Uranium. Since there are two types of fuel pins in the assembly, attention must be paid to the intra-fuel assembly power peaking factor and it should be kept to below 1.2. It was found that an equilibrium could be reached without violating any of the limits (> 5 w/o ^{235}U, peaking factor < 1.2) for this assembly type although at equilibrium an excess of Pu was produced due to the inner Uranium oxide (UOX) pins.[9] In this study equilibrium is considered to be obtained when k_{inf} varies by less than 10^{-6} at the end of a cycle for consecutive recycles.

2. Transuranics Multi-Recycling

The repository benefits as a result of recycling Pu as well as the minor actinides has also been considered. Removing Pu as well as Am from the waste can increase the capacity of a repository such as Yucca Mountain by a factor of five while also removing Cs and Sr will produce even greater benefits.[10,11] The Cs and Sr isotopes do not need to be transmuted due to there relatively short half-lives. Instead theses elements can be removed from the waste and stored separtely until they have decayed away.[10] There are many options for partitioning and transmuting the transuranics, from recycling only Pu and Np to recycling most of the transuranics in either a homogeneous or heterogeneous assembly.

Removing Neptunium as well as Plutonium from the waste has been considered to address proliferation concerns of separating only Pu. An added benefit to partitioning and transmuting the Np is that ^{237}Np does have an slight contribution the toxicity of the waste[12] and removing Np from the waste reduces the volume of waste (Np comprises about 6.5% of the transuranic waste[8]). Including Np is also beneficial for nonproliferation. The main isotope of Np is ^{237}Np which α decays to ^{233}Pa. ^{233}Pa has a half life of 27 days and emits a strong gamma with an energy of around 311

keV, resulting in the fuel being much hotter when Np is recycled. One downside to recycling Np is the required ^{235}U enrichment is increased for each cycle.[8] The final composition of the Pu+Np recycled fuel is not significantly different from that of a Pu-only recycle except that there is a increase in the concentration of ^{237}Np and ^{238}Pu.[13] The ^{238}Pu is the result of the neutron capture by a ^{237}Np isotope. There is a slight benefit for the Pu+Np case over the Pu-only case in terms of radiotoxicity and decay heat after about 10^6 years.[12] Overall recycling Pu+Np provides very little gain for a repository.

Unlike Neptunium, Americium contributes a significant amount to the toxicity of the spent fuel mostly due to the ^{241}Am isotope and this isotope is the primary source of ^{237}Np. Americium comprises about 5.5% of the total transuranics in the spent fuel waste and, along with Pu and Np, represents almost the entirety of the transuranic waste.[8] When recycled in a homogeneous fuel assembly the required enrichment of ^{235}U is nearly 6.5 w/o,[8] significantly greater than th 5 w/o limit for commercial reactors. When recycled in a heterogeneous assembly (UOX pins surrounded by TRU pins, no IMF) the required enrichment for ^{235}U is 5.5 w/o[12] and for a IMF heterogeneous assembly the required enrichment is 4.65 w/o.[9] When Americium is recycled, the amount of Curium and Californium is increased when compared to the previous two recycling strategies due to the neutron capture by the Americium isotopes.[13] Although the Curium and Californium isotopes are large neutron emitters as a result of spontaneous fission, they are also short lived. Because of their short half lives those two elements do not contribute significantly to the overall toxicity of the waste after about 100 years. The remaining toxicity from the waste in this recycling scenario is significantly lower than that of the previous scenarios discussed after about 100 years.[12]

So as to utilize the fissile Cm produced from recycling Am it has been proposed to

recycle this isotope as well. One mode of decay for the Curium isotopes is spontaneous fission, making the fabrication of a fuel with the inclusion of Cm difficult resulting in the need to remotely handle the fuel during this process. Because of the fissile properties of Cm the required enrichment or ^{235}U for each recycle is less than that of the previous recycle scenario.[8,12] When the fuel composition is at an equilibrium the concentrations of the Curium, Californium, and Berkelium isotopes are significantly larger than for the any other type of recycling strategy, requiring shielding during fabrication.[13] The toxicity of the waste as a result of this recycling strategy is the lowest of all scenarios considered after 10 years of cooling (once the short lived Cm, Bk, and Cf isotopes have decayed away).[12] The effects of recycling Berkelium and Californium were studied and it was found these isotopes provided no benefits since most had decayed away before the beginning of the next cycle and only posed a greater risk during fabrication.[8]

B. Increased Moderator-to-Fuel Ratio

With the addition of transuranics in the fuel, the neutron spectrum becomes hardened. A method to compensate for the reactivity loss due to the hardened spectrum is to increase the moderation. This can be done by either decreasing the pin diameter or increasing the rod pitch (i.e., distance between the fuel rods, removing fuel rods). When moderation is increased, a lower amount of Pu is required, resulting in a greater number of times that the Pu can be recycled before a positive void coefficient is reached.[14] It then must be determined how the additional moderation will be achieved, with an increased pitch or a reduced pin diameter. With either method the amount of fuel in the assembly will decrease, resulting in an economic penalty. Adjusting the moderator to fuel ratio (MFR) effects some of the reactor physics parameters such as fuel temperature coefficient (FTC) of reactivity, moderator temperature co-

efficient (MTC) of reactivity, or the moderator void coefficient (MVC) of reactivity. If the pin diameter is decreased the FTC increases slightly but remains negative.[14] The change is less drastic if the pin diameter is decreased rather than if the pitch is increased.[14] For a MOX fuel, MTC increases (becomes less negative) as the MFR is increased, while the MVC is inversely proportional to the MFR.[14]

C. Conclusions of Literature Survey

After the examination of the previous wok in topics discussed above, the following conclusions were drawn. To transmute the greatest amount of the UOX waste Pu, Np, Am, and perhaps Cm should be recycled in a homogenous assembly with increased moderation. The homogeneous assembly was chosen over a heterogeneous because the required ^{235}U and TRU enrichment could be kept at a low level and an equilibrium cycle was achieved in the homogeneous assembly studied by INL.[8] To increase moderation, we have chosen to reduce the pin diameter.

CHAPTER III

FUEL ASSEMBLY DESIGN AND RECYCLING SCENARIOS

This section presents the various fuel assembly designs as well as the parameters used to judge and compare the fuel assemblies. The assembly types that were developed for multi-recycling transuranics are an assembly design in which Plutonium, Neptunium, and Americium are recycled and a second design where the afore mentioned elements as well as Curium are recycled. These fuel types were then compared to a once-through UOX and MOX fuel assembly in terms of transuranics produced and the radiotoxicity of each design. A study of decreasing the fuel pin diameter as well as the effects on the centerline temperature is also presented. A reduced fuel diameter is necessary to compensate the loss of the thermal spectrum that is the result of recycling minor actinides. A smaller pin diameter will reduce the amount of fuel contained in a fuel assembly but the energy generated from the assembly should remain constant. As a result the fuel length is shortened when the pin diameter is reduced.

A. Overall Goals

The main objective of this Thesis is to simulate multi-recycling transuranics in a PWR fuel assembly and attempt at reaching a equilibrium in which the composition of the fuel does not change from cycle to cycle. If an equilibrium is achieved then the transuranics can remain in the fuel cycle indefinitely or at least until another transmutation solution is available. With the transuranics remaining in the fuel cycle the only waste that must be sent to a repository is the unusable fission products and the transuranic waste from reprocessing thereby significantly reducing the toxicity of the waste. It is customarily assumed that 0.1% of the transuranics end up in the

waste stream. A secondary benefit of recycling these transuranics is utilizing the fissile isotopes such as ^{239}Pu and ^{241}Pu that would otherwise be wasted. The use of these isotopes can reduce the amount of Uranium currently required to generate electricity.

B. Legacy UOX

The legacy UOX fuel is the spent fuel that currently exists throughout the United States. This spent fuel will be the source of transuranics for both the first generation of multi-recycled fuel assemblies as well as the supply of additional transuranics in subsequent generations. Using the legacy UOX fuel will deplete the stock pile of used fuel stored throughout the United States. The additional transuranics elements are used to make up for transuranics transmutation in the previous generation due to fission or losses during reprocessing. The legacy UOX is assumed to have been initially enriched to 3.86 wt% ^{235}U and depleted in a reactor to 45 GWd/tHM. The fuel is then assumed to have been cooled for an average of 20 years. The detailed isotopic concentrations at the beginning of cycle (BOC), end of cycle (EOC) and after 20 years of cooling are presented in Table III-I. The legacy fuel after 20 years of cooling is the sole supply of the transuranics in the first recycle generation. Since a significant amount of time would have passed between the beginning of the 1st and 15th recycle, the composition of the legacy would have changed. As can be seen in Table III-I the concentration of a few short lived actinides (^{238}Pu, ^{241}Pu, ^{242m}Am, and ^{244}Cm) would change significantly over a large period of time. As a result the effects of the decay of the legacy fuel was examined and is discussed in Section H of this chapter.

Table III-I. Isotopic concentrations for the legacy UOX fuel [atoms/(barn-cm)].

	BOC	EOC	EOC+20yr	EOC+150yr
^{234}U	1.278×10^{-6}	8.278×10^{-07}	1.649×10^{-06}	5.616×10^{-06}
^{235}U	9.078×10^{-4}	1.649×10^{-04}	1.650×10^{-04}	1.656×10^{-04}
^{238}U	2.232×10^{-02}	2.072×10^{-02}	2.072×10^{-02}	2.072×10^{-02}
^{237}Np	0.000	1.487×10^{-05}	1.536×10^{-05}	2.387×10^{-05}
^{238}Pu	0.000	6.527×10^{-06}	5.713×10^{-06}	1.751×10^{-06}
^{239}Pu	0.000	1.422×10^{-04}	1.445×10^{-04}	1.439×10^{-04}
^{240}Pu	0.000	6.521×10^{-05}	6.579×10^{-05}	6.552×10^{-05}
^{241}Pu	0.000	3.553×10^{-05}	1.568×10^{-05}	1.170×10^{-08}
^{242}Pu	0.000	1.923×10^{-05}	1.923×10^{-05}	1.923×10^{-05}
^{241}Am	0.000	6.793×10^{-06}	2.615×10^{-05}	3.332×10^{-05}
^{242m}Am	0.000	1.858×10^{-08}	1.709×10^{-08}	8.181×10^{-09}
^{243}Am	0.000	4.228×10^{-06}	4.227×10^{-06}	4.167×10^{-06}
^{244}Cm	0.000	1.473×10^{-06}	7.688×10^{-07}	2.470×10^{-09}
^{245}Cm	0.000	9.935×10^{-08}	9.921×10^{-08}	9.800×10^{-08}
^{246}Cm	0.000	1.178×10^{-08}	1.175×10^{-08}	1.149×10^{-08}
^{247}Cm	0.000	1.393×10^{-10}	1.393×10^{-10}	1.393×10^{-10}
^{248}Cm	0.000	9.334×10^{-12}	9.338×10^{-12}	9.335×10^{-12}

C. Energy Equivalence

To ensure that the recycled fuel assembly generates the same amount of energy as a UOX fuel assembly used in a current PWR, the concept of energy equivalence is developed. This concept is based on the linear reactivity model and assumes four cycles with a burnup of 15 GWd/tHM per cycle. To ensure energy equivalence, k_∞ at the average core burnup should be 1.0 with 0 ppm of soluble boron. With the assumption of a 3.5% neutron core leakage, the desired k_∞ becomes 1.035. Therefore, the enrichment of ^{235}U is altered each cycle to ensure that at a burnup of 37.5 GWd/tHM, k_∞ is 1.035. The value of 37.5 GWd/tHM represents the average burnup of a four-batch core depleted to 60 GWd/tHM. The four batches have been depleted to 15, 30, 45, and 60 GWd/tHM and the average core burnup is 37.5 GWd/tHM. To illustrate the concept of energy equivalence, Figure III-1 shows a comparison of k_∞ as a function of burnup for a 4.9 w/o UOX assembly and three first-generation TRU assembly with Cm. The three TRU assemblies contain 8 weight percent transuranic elements and three different enrichments of ^{235}U. The TRU assembly with a ^{235}U enrichment of 3 w/o is considered to generate the same amount of energy as the UOX assembly since the value for k_∞ at 37.5 GWd/tHM is 1.035 for both assembly types.

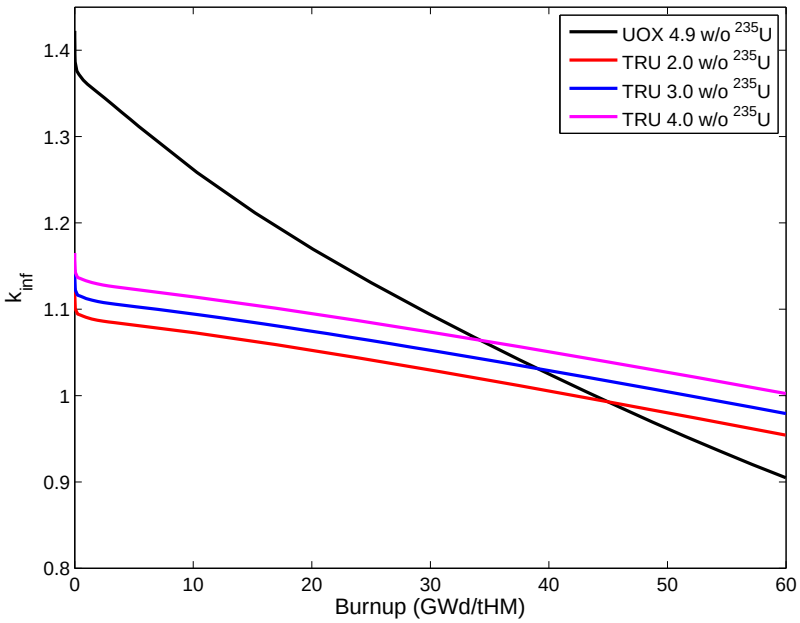


Fig. III-1. Energy equivalence analysis between a 4.9w/o UOX and 3 TRU+Cm fuel assemblies.

D. Voiding Analysis

One of the safety factors that will be considered in this Thesis is the void reactivity coefficient which should remain negative for a reactor to be inherently safe. If the void coefficient was positive then an increase in the void will result in a reactivity increase causing a increase in power. This in turn would cause a increase in the void fraction resulting in a further reactivity increase. A negative void coefficient will result in a decrease in the reactivity and subsequently reactor power resulting in a lower void fraction. Reactivity is considered to have decreased when the value of k_∞ for an assembly with a reduced moderator density is less than the value of k_∞ for nominal moderator density.

The source of the additional reactivity when an assembly is voided is the result

of an increased fission rate from fast fissions. Isotopes such as ^{238}U, ^{240}Pu, and ^{242}Pu capture neutrons with thermal and epithermal energies while fast neutrons can induce fission for these isotopes. When the fuel is voided the moderator is lost and the neutrons produced remain fast. For a standard UOX fuel the only isotope mentioned above of interest is ^{238}U. A hardening of the neutron spectrum will result in an increase in the number of fast fission by ^{238}U but the increased absorption rate at fast energies by ^{238}U will compensate for the increased fast fission rate. Because of this ^{238}U provides a negative contribution to the void coefficient.[15] This can be seen in Figure III-2 where as the ratio of moderator to fuel atoms decreases (increased void fraction) the reactivity of the fuel decreases.

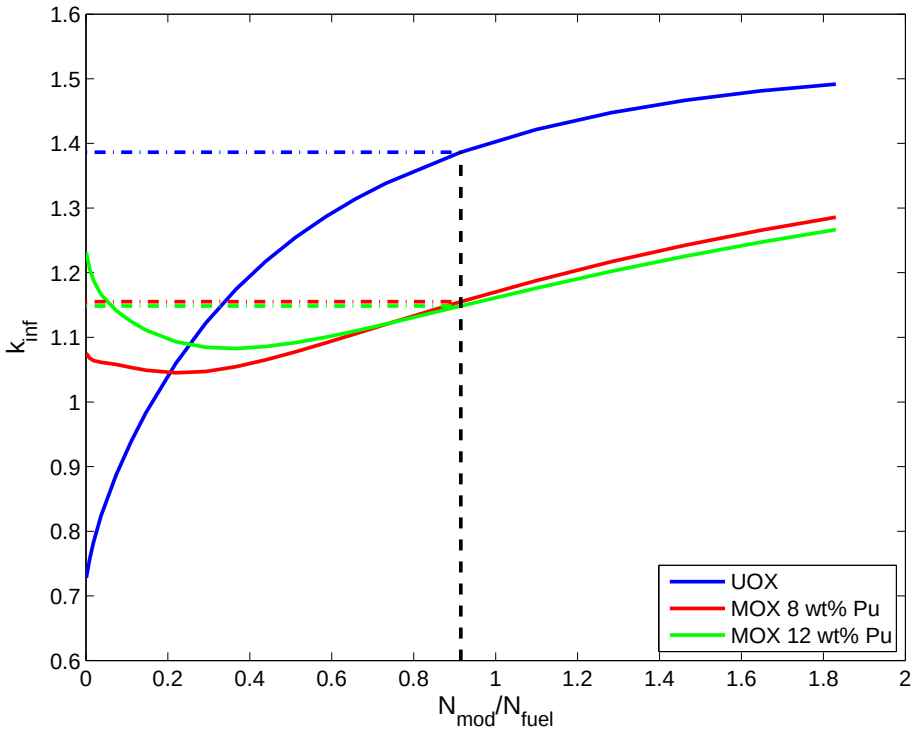


Fig. III-2. k_∞ as a function of moderator to fuel ratio for a UOX and two MOX assemblies.

The reactivity of the MOX fuel increases for high void fractions as a result of the addition of Plutonium. The fission probability of the even Plutonium isotopes (^{240}Pu and ^{242}Pu) increases as the neutron energy increases; this is the cause of the reactivity increase for higher void fractions. The additional Plutonium in the MOX fuel reduces the amount of ^{238}U which, as discussed previously, contributes to the negative void coefficient. Therefore, an increase in the Plutonium content penalizes the void coefficient in two ways; an increase in the concentration of even Plutonium isotopes and a decrease in the concentration of ^{238}U. As a result, a limit exists in the amount of Plutonium that can be contained in the fuel before the reactivity of the fuel will increase. In Figure III-2, the UOX and MOX assemblies with 8 w/o Pu have a negative void coefficient while the MOX assembly with 12 w/o Pu has a positive void coefficient.

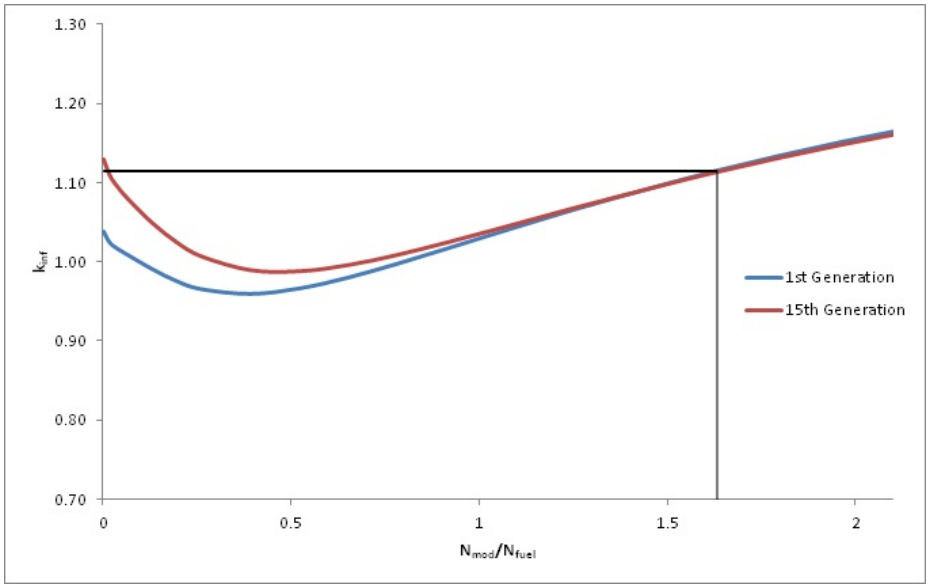


Fig. III-3. Voiding analysis for two generations of multi-recycled fuel assemblies.

An example of the voiding analysis that will be performed for all fuel assembly

designs can be seen in Figure III-3. In this figure a voiding analysis was performed for a first generation and a fifteenth generation multi-recycled transuranic pin without Cm. The weight percent of TRU in both fuel assemblies is 8 w/o and the ^{235}U enrichment is 3.02 w/o for the first generation assembly and 5.64 w/o for the 15$^{\text{th}}$ generation assembly. For a standard moderator density, k_∞ for the first and fifteenth generation at beginning of irradiation is 1.11 while for a voided condition (low N_{mod}/N_{fuel} values) the reactivity of both fuel designs begins to increase. The black horizontal line in the figure represents the max value for k_∞ in a voided situation while the vertical line represents the standard ratio for N_{mod}/N_{fuel}. The value for a the standard moderator to fuel ratio has shifted for the reduced fuel pin is greater since there is less fuel and more moderator when the pin is reduced. The value for k_∞ when assembly is completely voided for the first generation remains below the value for k_∞ for standard moderation. This is not the case for the 15$^{\text{th}}$ generation fuel assembly. The value of k_∞ in voided conditions is greater than k_∞ for standard conditions. Based on the voiding analysis the fifteenth cycle would not be acceptable while the first cycle would pass this requirement.

E. Transmutation

In this study, one of the purposes is to determine the ability of a PWR to transmute transuranic elements so as to reduce the amount of waste generated in the nuclear fuel cycle. Through the absorption of a neutron a fertile isotope can be transmuted to a fissile isotope which will then fission producing short lived fission products that can easily be stored. The drawback is that not every neutron absorption results in the production of a fissile isotope. Instead the absorption of a neutron by a transuranic isotope could result in the production of a heavier isotope. This is a concern when the fuel already contains some heavy actinides. The inclusion of Curium in the fuel will

result in a greater production of Berkelium and Californium both of which have short half-lives and will contribute significantly to fuel fabrication hazards. This concern is greater for fuel containing Curium than those where it is not recycled because when Cm is left in the waste, then an isotope must be transmuted through the entire Curium vector (^{244}Cm - ^{248}Cm) before a short lived Curium isotope is found. ^{249}Cm is the first isotope on the Curium vector that does not α-decay and has a short enough half-life that it is less likely this isotope will be transmuted prior to decaying. This isotope is the gateway to heavier isotopes such as ^{249}Cf since it will more likely β-decay ($t_{1/2}$=64.15 min.) to ^{249}Bk before continuing the Cm chain and absorbing absorb another neutron. If Cm is recycled then ^{248}Cm will exist initially and ^{249}Cf will be produced more rapidly.

The rate of transuranic production or consumption will serve as a metric by which any recycling strategy can be compared to one another. During each fuel cycle, ^{238}U will be transmuted to heavier isotopes while some of the transuranics included initially will fission. A beneficial recycling scenario will destroy a greater amount of transuranic isotopes than it creates. The amount of TRU produced or consumed will be determined using Eq. 3.1 where m_{EOC} is the mass at the end of the cycle (after five years cooling) and m_{BOC} is the mass at the beginning of the cycle (prior to the two year cooling).

$$\text{TRU Production}(\%) = \frac{m_{\text{EOC}} - m_{\text{BOC}}}{m_{\text{BOC}}} \tag{3.1}$$

F. Description of the Fuel Assembly

The fuel assemblies modeled in this Thesis are based on a typical Westinghouse PWR 17x17 assembly. It was later determined that additional moderation was required and the diameter of the pin was reduced as will be discussed in a later section. Table III-

II[16] contains the dimensions of a Westinghouse PWR fuel assembly.

Table III-II. Dimensions of a typical Westinghouse 17x17 PWR fuel assembly.[16]

Assembly Lattice	17x17
Number of Fuel Pins	264
Number of Guide Tubes	24
Number of Instrumentation Tubes	1
Fuel Rod Pitch	1.26
Inter-assembly Gap	0.08
Fuel Material	UOX
Cladding Material	Zircaloy-4
Fuel Pellet Radius (cm)	0.4096
Gap Thickness (cm)	0.0084
Cladding Thickness (cm)	0.057
Discharge Burnup (GWd/tHM)	60.0
Fuel Temperature (K)	900.0
Cladding Temperature (K)	581.0
Nominal Coolant Density (g/cc)	0.72

Only a homogeneous fuel assembly will be considered for recycling the transuranics. All of the UOX pins in a standard PWR assembly will be replaced with a fuel pin containing recycled transuranics (TRU pins). Each of the TRU pins will contain 8 w/o transuranics so as to prevent a positive void coefficient. If a fuel type does have a positive void coefficient then that assembly is considered to have failed. The degradation of the fissile content of the recycled transuranics will be compensated by

increasing the enrichment of Uranium so as to achieve the desired energy equivalence. The maximum ^{235}U enrichment at any recycle is 5 w/o, the legal limit for commercial fuel in the United States. The necessary Uranium enrichment is determined by respecting the energy equivalence between fuel assembly designs.

G. Increased Moderation

1. Reduced Fuel Pin Diameter

One of the results of adding transuranics to the fuel is that the absorption rate of the fuel increases, resulting in a hardening of the neutron spectrum. Because of the hardened spectrum, there are fewer thermal neutrons to induce fission and sustain a chain reaction. To compensate for the loss of thermal neutrons and subsequently the reactivity loss, the enrichment of fissile nuclides could be increased. The only source of additional fissile nuclides that could be added to the fuel would be to increase the enrichment of ^{235}U since increasing the transuranics content could produce a positive void coefficient. It was quickly found that increasing the enrichment of Uranium would not be a viable solution as the limit on ^{235}U enrichment was exceeded after only one cycle.

To offset the increased absorption by the transuranics the amount of moderation could be increased. To increase the moderation the diameter of the fuel pin will be reduced although another option is to increase the fuel rod pitch. A study was performed to determine what effect decreasing the fuel pin radius has on the required enrichment to maintain energy equivalence. The fuel is composed of 8 w/o transuranics including Cm and burned to 60 GWd/tHM. The cladding thickness was kept constant for each case. In the study an arbitrary ^{235}U enrichment was chosen and the pin radius was altered so as to achieve energy equivalence. The core power

will remain constant in each case but the amount of fuel required to provide this thermal power will change as the radius is altered, therefore affecting the power density. Power density is defined as the total core power divided by the mass of the fuel (W/g). As the radius of the fuel pin is reduced, the amount of fuel decreases and the power density increases. Burnup is then the power density multiplied by the amount of time the fuel is being irradiated (campaign length). Since the burnup will remain constant and the power density will increase with a decreasing radius, it follows that the campaign length will be shortened. Table III-III contains the results of the fuel pin radius study. As can be seen, reducing the pin radius also reduces the campaign length. This is a necessary trade-off because without additional moderation the transuranics could not be multi-recycled. The pin radius that will be used hereafter is 0.3557 cm, a reduction of 13.8% from the standard fuel pin radius. This new pin radius will shorten the campaign length by 407 days. Figure III-4 compares the difference between a standard PWR fuel pin (dashed line) and the reduced fuel pin (red circle). Figure III-5 is a comparison of the neutron spectrum for a standard PWR fuel pin and the reduced fuel pin that will be used in this study. The result of the additional moderation can easily be seen as the population of thermal neutrons increases as for the reduced pin diameter which is necessary to reduce the required ^{235}U enrichment.

Table III-III. Required pin radius to achieve energy equivalence as well as power density and campaign length for various ^{235}U enrichments.

^{235}U Enrichment (wt%)	Pin Radius (cm)	PD (kW/kg)	Campaign Length (days)
2.5	0.3355	53.66	1119
3.0	0.3557	47.75	1257
3.5	0.3720	43.66	1374
4.0	0.3850	40.76	1472
4.5	0.3990	37.95	1580
5.05	0.4127 (nom.)	35.48	1664

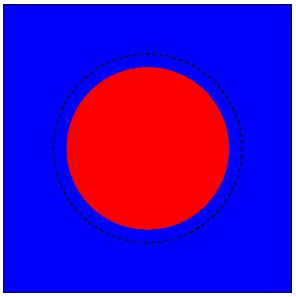


Fig. III-4. Comparison between a standard PWR fuel pin (dashed line) and the reduced diameter fuel pin (red circle).

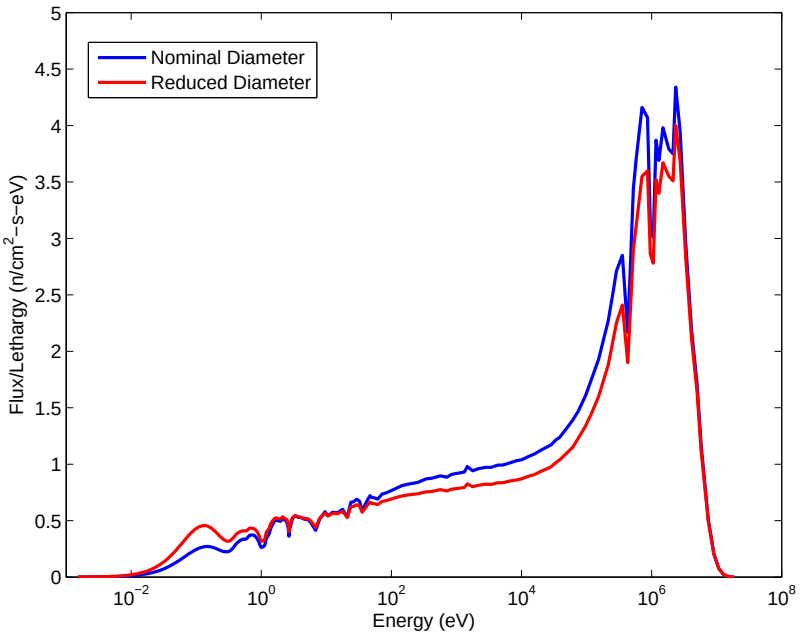
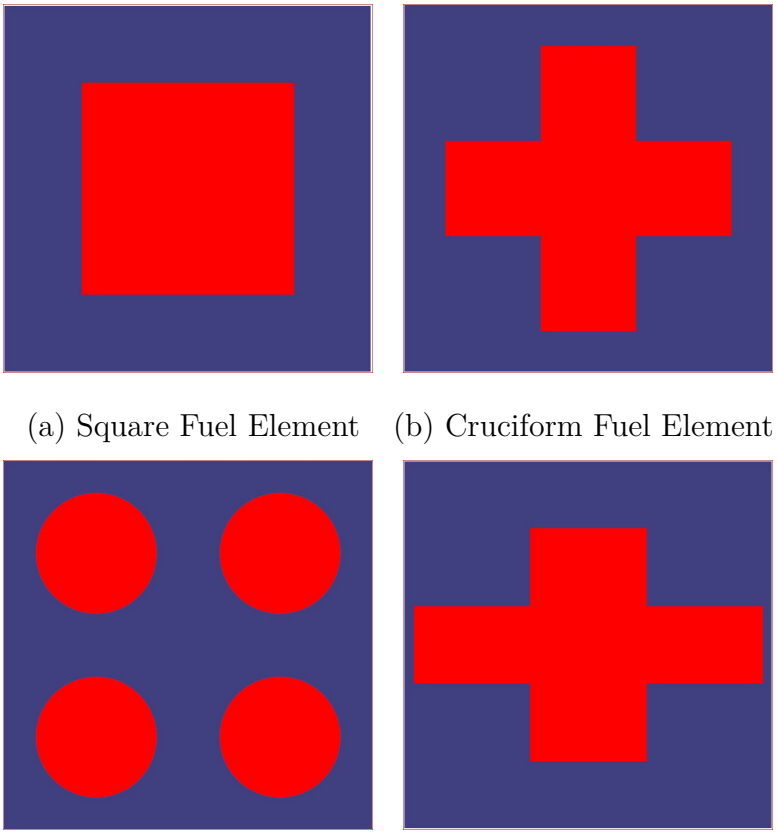


Fig. III-5. Comparison of the neutron spectrum for a standard PWR and reduced diameter fuel pin.

2. Other Geometries Examined

So as to not sacrifice campaign length for additional moderation other geometries and configurations were considered. Four types of geometries were examined, a cruciform with arms of equal sizes, a cruciform with arms of unequal sizes, a square pin, and four smaller pins replacing one normal sized pin. The fuel volume remained constant in each of the examined geometries (i.e., the fuel volume of the four fuel pins equals the volume of one standard pin) so that the campaign length would remain constant. The four geometries examined are shown in Fig. III-6.



(a) Square Fuel Element (b) Cruciform Fuel Element

(c) 4 Fuel Pins (d) Uneven Cruciform Fuel Element

Fig. III-6. Geometries examined to obtain additional moderation.

In this study the effects of the cladding is neglected leaving only the fuel and moderator being modeled. For each geometry the fuel pin is composed of a MOX fuel containing 8 w/o Pu and 3 w/o ^{235}U. Table III-IV presents the initial multiplication factor for each of the geometry types. The best geometry is the cylindrical fuel pellet. Knowing this, the pin radius that will be used in the study will be of a reduced diameter for the additional moderation.

Table III-IV. Initial multiplication factor for the four geometries examined as well as a standard fuel pellet.

Geometry	k_∞
Pellet	1.20493
Square	1.19794
Cruciform	1.18444
Uneven Cruciform	1.18538
Four Pins	1.14702

H. Radiotoxicity

The purpose of multi-recycling the legacy UOX fuel is two-fold, first to utilize the fissile and fertile isotopes contained in the used fuel and secondly to reduce the radiotoxicity of the waste from the nuclear fuel cycle. Radiotoxicity is not a measure of the activity of a radionuclide but instead more aptly described as the hazard a given radionuclide poses to the human body if it is inhaled or ingested. For the purpose of this study, radiotoxicity is the volume of air or water required to dilute a given amount of radionuclide to acceptable limits.[17] Since the waste will eventually be placed in long term storage, ingestion toxicity is of the greatest concern: the waste could leak from the repository into the underground aquifer. As a result of this, the radiotoxicity that will be considered for this study is the amount of water required to dilute a given amount of radionuclides to the permitted limit.

1. Decay Adjustment

With the length of each cycle being approximately 10.5 years (2 year fabrication + 3.5 years irradiation + 5 years cooling), the shorter lived isotopes in the waste from the first cycle will have decayed away and may need to be taken into account. Most of the short lived isotopes are fission products. Figure III-7 shows how much of the fuel toxicity is the result of fission products and how much is attributed to the actinides. As can be seen, the fission products result in a significant portion of the toxicity initially but after 100 years are negligible. Because each fuel type will be recycled for 15 cycles the waste from the first cycle would have decayed for 147 years before the completion of the 15^{th} cycle. As a result the short lived fission products which contribute significantly to the radiotoxicity of the fuel in the first 100 years would have completely decayed away by the completion of the 15^{th} cycle. Instead of determining what the composition of the waste from each cycle will be at the completion of the 15^{th}, if is assumed that the radiotoxicity of fuel for the first 100 years after the 15^{th} cycle is conservative but after that point, the values are accurate.

The assumption above accounts for the short lived fission products but neglects the decay of actinides and does not address the effect of the short-lived actinides (i.e., ^{241}Pu, ^{242m}Am, ^{243}Cm, and ^{244}Cm). Prior to the completion of the final recycle, only 0.1% of the actinides recycled are included in the waste. Following the discharge of the 15^{th} cycle, the population of theses recycled isotopes increases by three orders of magnitude. As a result, the impact of the decay of the short-lived isotopes is insignificant, as can be seen in Figure III-8. In the figure, the toxicity of the waste when the decay of ^{238}Pu, ^{241}Pu, ^{243}Cm, and ^{244}Cm is accounted for is represented by the red line, while the toxicity of the waste when the decay of these isotopes is neglected is represented by the blue line. As can be seen there is no difference in the

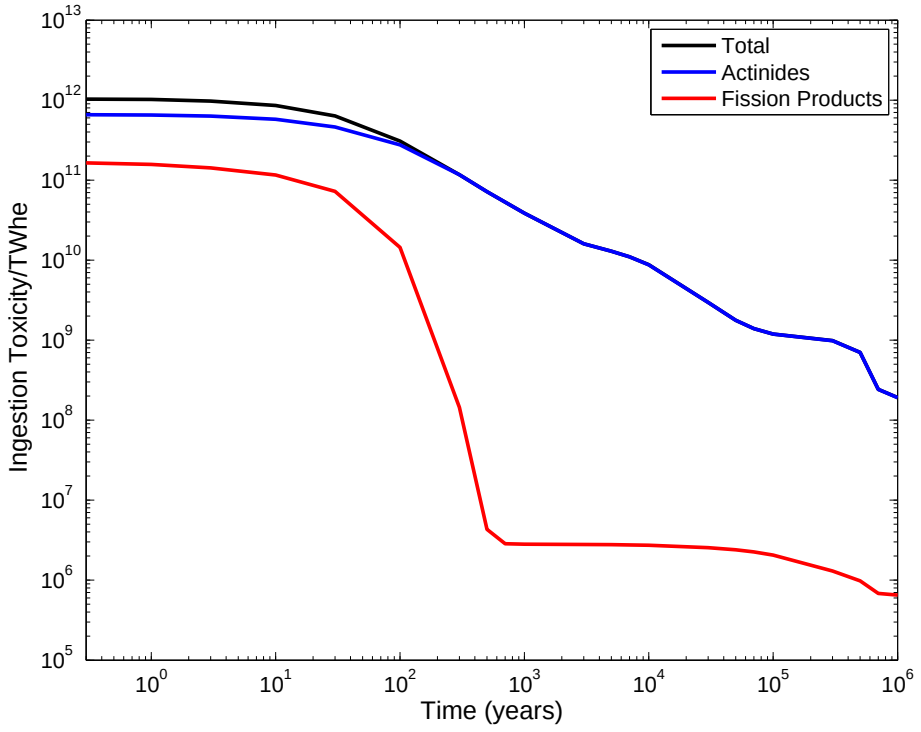


Fig. III-7. Comparison of the contribution to the total toxicity value for the actinides and fission products.

toxicity of the waste when the decay of these isotopes is accounted for and need not be done.

The decay of legacy UOX fuel was also considered. As seen in Table III-I there is a significant decrease in the concentration of ^{238}Pu, ^{241}Pu, ^{242m}Am, and ^{244}Cm. Of these isotopes, the loss of ^{241}Pu affects the reactivity of the fuel the greatest. As a result, the enrichment of ^{235}U must be increased. A study of the required enrichment of ^{235}U for the 15^{th} recycle using 20 year cooled and 150 year cooled legacy UOX found that enrichment increased from 4.79 wt% to 4.98 wt% respectively. The change in the radiotoxicity of the fuel was also examined. Figure III-9, compares the radiotoxicity of the fuel when a 20-year cooled and 150-year cooled legacy fuel is used to create

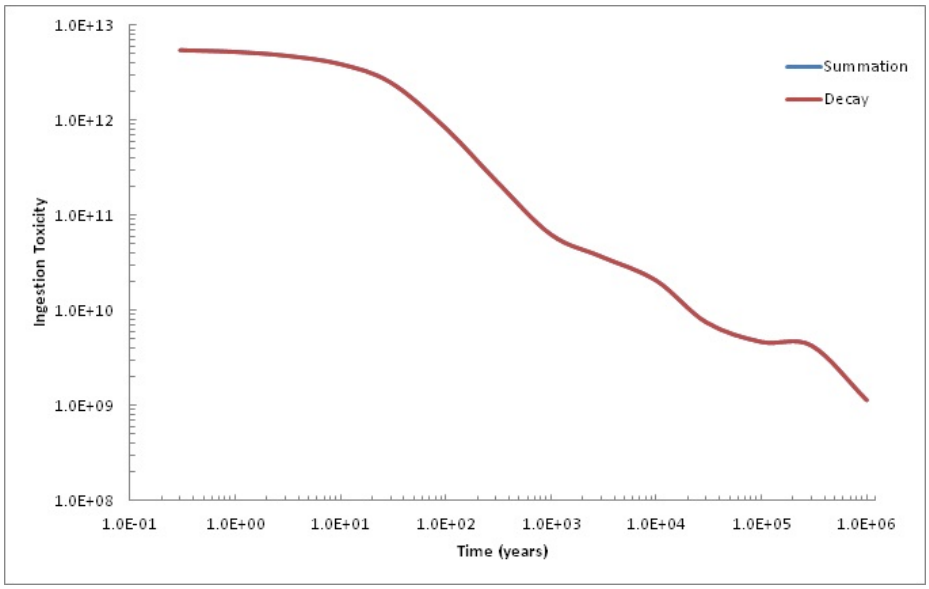


Fig. III-8. Comparison of the toxicity of the actinides when accounting for the decay of some short lived actinides and when decay is neglected.

the 15th recycle. In the figure, only the radiotoxicity of the 15th recycle fuel was considered, not any of the waste from the previous recycles.

As can be seen in Figure III-9, the difference between the radiotoxicity using the two legacy sources is not significant. One must also remember that the legacy UOX stockpile is continuously being replenished as UOX fuel is continued to be used in the United States. As a result the loss of ^{241}Pu will not be as drastic as was assumed in this study. As a result the legacy fuel that will be used is a legacy UOX fuel that has been cooled on average for 20 years.

2. Radiotoxicity Normalization

To compare the waste from a once-through cycle and a multi-cycle fuel, the radiotoxicity of the fuel in question will be normalized to the amount of electricity produced. After the 15$^{\text{th}}$ cycle the entire content of the fuel assembly is assumed to be discharged

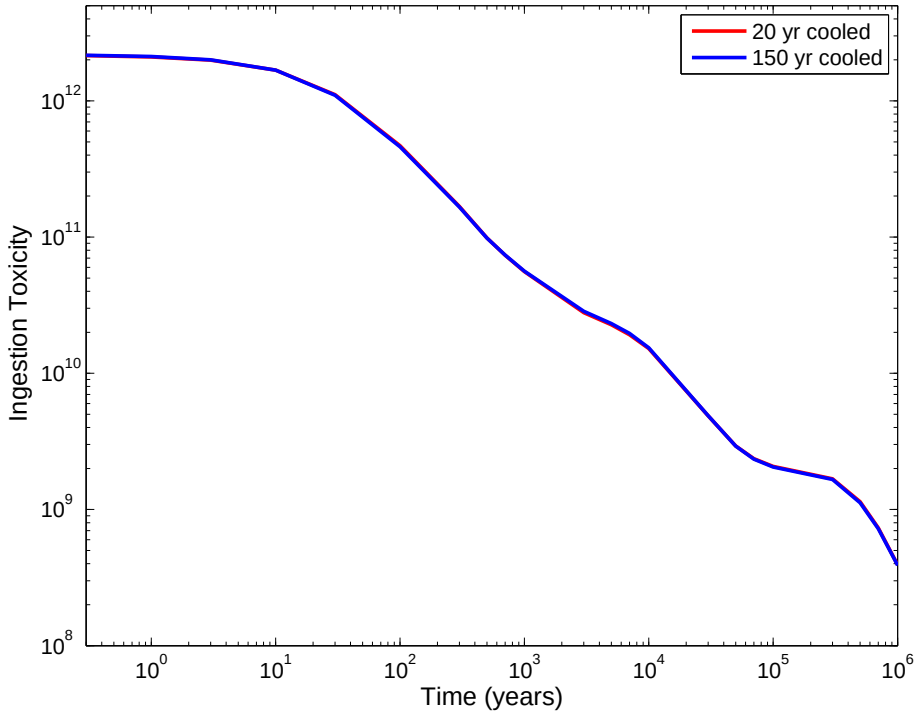


Fig. III-9. Comparison of the toxicity for the 15th recycle when a 20 year cooled and 150 year cooled legacy UOX is used.

to waste. Since TRU from legacy UOX are being used to make up for the losses from one cycle to another, legacy UOX must be appropriately included in the toxicity of the recycled fuel. Eq. 3.2 is used to normalize each isotope in the final waste stream based on the electricity generated.

$$\text{Normalized Isotope Value}\left(\frac{\text{moles}}{\text{TWhe}}\right) = \frac{\sum_{i=1}^{N_{\text{cycles}}}\left(C_i + n_i \times C_L\right)}{\sum_{i=1}^{N_{\text{cycles}}}\left(E_i + n_i \times E_L\right)} \tag{3.2}$$

where:

C_i is the amount of waste from cycle i in moles

n_i is the number of legacy UOX assemblies utilized to ma up fuel for the current

cycle, i

L is the legacy UOX waste remaining following the removal of the desired actinides in moles

E_i is the power generated by the recycled assembly in cycle i, 0.196 TWhe

E_L is the power generated by the legacy UOX assembly, 0.205 TWhe

The value of C_i, n_i will be different for each fuel cycle scenario while the energy produced by the recycled fuel, E_i, will not change with each recycle.

3. Natural Uranium Toxicity

Each fuel cycle will be compared to the toxicity of natural Uranium (Unat). Figure III-10 is a plot of the toxicity of 1 kg of Unat as a function of time. Also included in the figure is the toxicity of some of the significant daughter products. The composition of Unat is considered to be 0.72% ^{235}U, 5.5×10^{-3}% ^{234}U, with the remainder being ^{238}U. Of course this, is not not representative of the Unat ore today since it neglects the presence (and contribution to radiotoxicity) of many daughter elements. This issue is resolved as follows: Initially the radiotoxicity of Unat is combination of the toxicity of ^{234}U and ^{238}U with ^{235}U not being a significant contributor. After about 1,000 years the toxicity of Unat begins to increase before reaching a plateau at a higher value where it remains for an extended period of time. This increase is the result of the build up of daughter products as the result of the decay of the Uranium atoms. The daughter products presented in Figure III-10 are those that contribute the most to the total radiotoxicity of Unat, not those that build up to the greatest quantity. It takes about 1,000 years before these daughter products contribute significantly to the total radiotoxicity but after this point they are the main portion of the total

toxicity. The Uranium that is found on Earth has been decaying for billions of years and has reached a transient equilibrium with the daughter products. A transient equilibrium occurs when the half -life of the parent is greater than the half-life of the daughter product. As a result the activity of the daughter nuclides will reach a maximum value before decreasing at the same rate as the parent activity.[18] Since this equilibrium exists currently with the Uranium and its daughter nuclides, it is assumed that the toxicity has reached the plateau seen in Figure III-10 and this value will be used as the toxicity of Unat. Using ORIGEN it was found that 4.65×10^{4} cubic meters of water is required to dilute 1 kg of natural Uranium below permissible levels.

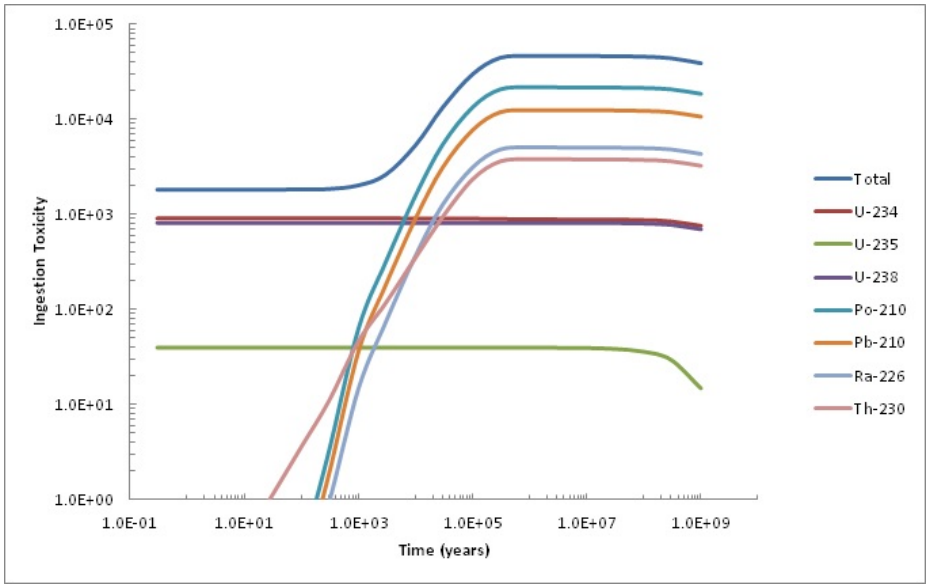


Fig. III-10. Toxicity of natural Uranium and significant daughter products.

The radiotoxicity value found above is for 1 kg of Unat and next must be converted to the radiotoxicity of the Unat used in a given fuel cycle. To do this the amount of natural Uranium used in each fuel cycle must be determined. For a UOX

direct cycle the process is straightforward. The radiotoxicity of the natural Uranium used in a UOX fuel assembly is found using Eq. 3.3. In the equation, the radiotoxicity has been normalized to the electricity produced by a UOX assembly so as to compare to the normalized radiotoxicity of the UOX assembly.

$$T_{nat}^{UOX} = \frac{T_{nat} \cdot F/P \cdot U}{E_{UOX}} \tag{3.3}$$

where:

T_{nat}^{UOX} is the radiotoxicity of Unat required to create one UOX assembly

T_{nat} is the radiotoxicity of 1 kg of Unat

F/P is the feed to product ratio found using Eq. 3.4

U is the mass (kg) of Uranium in a UOX assembly

E_{UOX} is the electricity produced by the UOX assembly, 0.263 TWhe.

In the above equation, the variable F/P is a ratio of the mass of the feed (natural Uranium) to the mass product (enriched Uranium). It is a unit-less value that, when multiplied by the amount of Uranium in a fuel assembly, determines how much natural Uranium must go through the enrichment process to produce 1 fuel assembly. The by-product of the enrichment process is depleted Uranium, commonly refereed to as the tail. The ^{235}U weight percent of the tail is usually about 0.2 w/o and this value used is in this study. The feed to product ratio is found using Eq. 3.4 where x_p is the weight percent of the product, x_f is the weight percent of the feed, and x_t is the weight percent of the tail. Table III-V presents the feed to product ratio for the two multi-recycled fuels. In the table TRU+Cm is the fuel type where Cm is recycled while TRU-Cm is the fuel type where Cm is not recycled.

$$F/P = \frac{x_p - x_t}{x_f - x_t} \tag{3.4}$$

Table III-V. Feed to product ratio for each fuel type for each recycle.

Cycle	TRU+Cm	TRU-Cm
1	5.479	5.519
2	8.121	8.444
3	8.865	9.511
4	9.041	9.951
5	9.061	10.176
6	9.061	10.313
7	9.041	10.403
8	9.041	10.470
9	9.022	10.519
10	9.022	10.560
11	9.022	10.591
12	9.002	10.611
13	9.002	10.628
14	9.002	10.642
15	8.982	10.652

Eq. 3.3 is applicable for determining the radiotoxicity of the required natural Uranium for a direct cycle but not for a recycled assembly. When a fuel assembly is recycled, one must account for the natural Uranium required to create the recycled assembly as well as the natural Uranium used to create the legacy UOX assemblies. To determine the radiotoxicity of natural Uranium used throughout the multiple

fuel cycles, Eq. 3.5 will be used. The value found will then be compared to the radiotoxicity of the multi-recycled fuel to determine when the waste from the fuel falls below that of natural Uranium. Eq. 3.5 can also be used to determine the radiotoxicity of natural Uranium for a MOX fuel assembly. The only difference is that there would only be one cycle eliminating the need for the summations.

$$T_{nat}^{recycled} = \frac{T_{nat}\left[\sum_{i=1}^{N_{cycles}} \left(F/P_i \cdot U_i + F/P_L \cdot n_i \cdot U_L\right)\right]}{\sum_{i=1}^{N_{cycles}} \left(E_i + n_i \cdot E_L\right)} \tag{3.5}$$

where:

$T_{nat}^{recycled}$ is the radiotoxicity of Unat used for the entire fuel cycle

N_{cycles} is the number of recycles performed during the fuel cycle

T_{nat} is the radiotoxicity of 1 kg of Unat

F/P_i is the feed to product ratio for the recycled fuel assembly

U_i is the mass (kg) of Uranium in the recycled fuel assembly

F/P_L is the feed to product ratio for legacy UOX assembly

U_L is the mass (kg) of Uranium in the legacy UOX assembly

n_i is the number of legacy UOX assemblies used in the recycled fuel assembly

E_i is the electricity produced by the recycled assembly

E_L is the electricity produced by the legacy UOX assembly

I. Recycling Strategies

Two recycling strategies will be examined: the recycling of Pu, Np, and Am and the recycling of Pu, Np, Am, and Cm. The methods described will apply to both types of fuel. The initial loading of the proposed fuel will compose of TRU from legacy UOX. The initial loading in the recycled fuel will be 8 w/o TRU with the rest being enriched Uranium (UE). The ^{235}U enrichment will be adjusted so as to achieve energy equivalence as described earlier.

Once the composition of the fuel assembly has been determined, a two-year decay period is applied to account for the time required for fabrication and transportation. Next, the fuel will be irradiated in the reactor core for a burnup of 60 GWd/tHM and will then rest in the spent fuel pool to decay for 5 years. This will signify the completion of the first cycle. The used fuel will then go through reprocessing where the Pu, Np, Am, and, depending on the recycling scenario, Cm will be removed, with an assumed 99.9% efficiency. The remaining isotopes, including the 0.1% TRU that is not recovered, is considered the waste from the first cycle. The second cycle then begins with the isotopes removed from the previous cycle and legacy UOX is added to increase the TRU enrichment back to the 8 w/o and additional UE is added to achieve energy equivalence. This process is then repeated for multiple cycles. Figure III-11 summarizes the process followed for a recycling strategy.

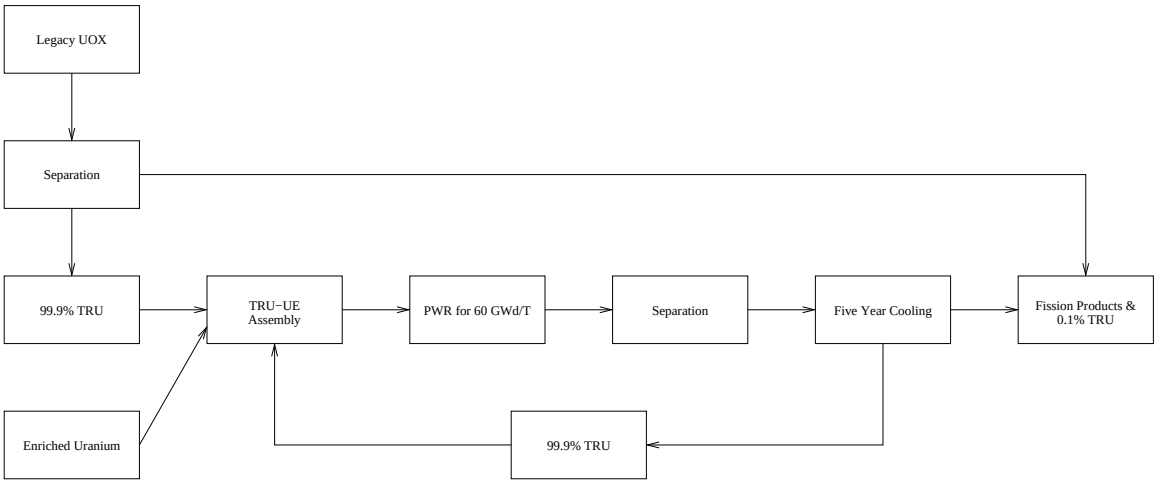


Fig. III-11. Transuranics multi-recycling strategy.

CHAPTER IV

TECHNICAL APPROACH

The research performed in this Thesis is based solely on computer simulations. A lattice physics code (DRAGON) to simulate fuel burnup and an isotopic decay package (ORIGEN) with the ability to determine radiotoxicity are employed to perform these simulations and will be described in the sections that follow. A description of the methodology followed to analyze a fuel assembly is also presented in this chapter. Figure IV-1 summarizes the process followed for each fuel recycle.

A. Design Procedure

1. Fuel Fabrication

To begin each fuel cycle the composition of the fuel assembly must be determined. The fuel is comprised of TRU from the legacy UOX (described in a previous section), enriched Uranium and, after the first cycle, the isotopes recycled from the previous cycle. The mass of recycled material is determined using Eq 4.1. In Eq 4.1, m_{k-1} is the mass of the recycled material, m_i is the mass of an individual recycled isotope, n is the number of isotopes recycled, N_i^{k-1} is the number density of a recycled isotope in units of atoms/b-cm, M_i is the atomic mass of a given isotope, N_a is Avogadro's number, r is the radius of the fuel pin and h is the length of the fuel pin.

$$m_{k-1} = \sum_{i=1}^{n} m_i = \sum_{i=1}^{N} \frac{N_i^{k-1} M_i}{N_a} \pi r^2 h \tag{4.1}$$

To determine the mass of the transuranics in the next recycle Eq 4.2 is used.

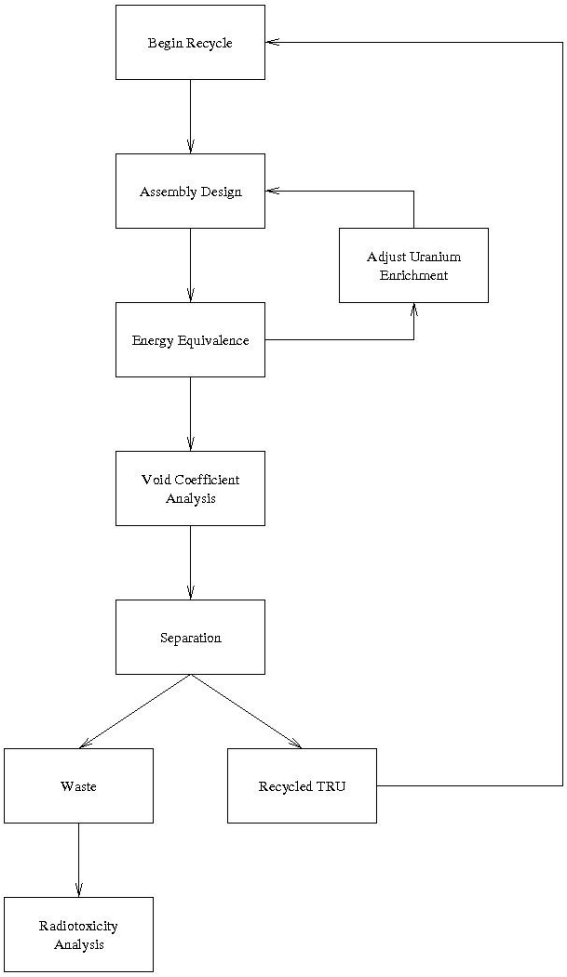


Fig. IV-1. Fuel recycle process.

$$m_k^{TRU} = m_{k-1}^{TRU} + m_{Legacy} \tag{4.2}$$

Once the mass of the transuranics is known, the mass of the Uranium is then determined. The weight percent of transuranics in the fuel is an input variable and by dividing the mass of the transuranics by this weight percent the total mass of metal in the fuel is known. The mass of Uranium in the fuel then becomes the total metal fuel mass subtracted by the mass of the transuranics. Equation 4.3 summarizes how

the mass of Uranium is found. In this equation m_U is the mass of uranium, m_{TRU} is the mass of the transuranics, and w_{TRU} is the weight percent of the transuranics in the fuel.

$$m_U = \frac{m_{TRU}}{w_{TRU}} - m_{TRU} \tag{4.3}$$

The amount of Oxygen contained in the fuel element is simply the sum of the number of heavy metal atoms (U, Pu, Np, Am, and Cm if any) multiplied by 2 since there is two oxygen for each heavy metal (i.e., UO_2). The number of metal atoms is found multiplying the density of the fuel by Avogadro's and dividing by the atomic mass. With the composition of the fuel for the current cycle known the power density is then found by dividing the mass of heavy atoms by the thermal power produced which was chosen to be is 3800 MW_{th} for a typical 193 fuel assembly PWR core.

2. Irradiation Simulation

A simple text file is created (using Excel) containing the fuel composition and density, power density, irradiation length, and pellet dimensions. A Perl script is used to extract the information from the text file and create an input deck for DRAGON. DRAGON then computes the decay that occurs during the two-year fabrication and transportation phase, the irradiation to 60 GWd/tHM, and the decay during the five-year cooling period. Following the DRAGON run, an additional Perl will gather information from the output file such as k_{inf} as a function of burnup and the final fuel composition. It will then be verified that fuel assembly achieved an energy equivalence with a standard UOX assembly as previously discussed. If not the case, then the ^{235}U enrichment will be altered as necessary and the previous steps will be repeated. If energy equivalence is achieved, then a voiding analysis is performed to ensure the

void coefficient remains negative. If the fuel cycle passes both of these checks, then the actinides of interest are extracted from the final fuel composition file. A recycling efficiency of 99.9% is assumed and applied to the actinides of interest, resulting in 0.1% of the actinides going into the waste stream. The waste from the recycling process as well as the isotopes that are not recycled are then stored separately to make up the composition of the final waste. This process is repeated for 15 cycles or until an isotopic equilibrium is reached.

3. Radiotoxicity

Following the completion of the 15^{th} cycle, the radiotoxicity of the waste will be determined. First the ingestion toxicity of a UOX direct cycle, a once through MOX cycle, and the two multi-recycled fuel cycles will be determined. Using a Perl script, the concentration (atoms/b-cm) of discharged UOX fuel will be extracted from the DRAGON output file and stored in a separate file. Theses concentrations will be converted to the number of moles (input units to ORIGEN) of each isotope present in the entire fuel assembly and normalized to the amount of electricity generated by the UOX assembly. The resulting values have units of moles/TWhe. For the MOX, the concentrations will be extracted from the DRAGON output and converted to moles. Then using Eq. 3.2 the discharged MOX fuel will be combined with the legacy UOX assembly waste and normalized based on the electricity generated by the MOX and legacy UOX assemblies. The legacy UOX waste that will be used to determine the MOX radiotoxicity will only contain 0.1% of the Pu isotopes since these isotopes were removed from the legacy UOX stockpile and used to create the MOX fuel. This process will then be repeated for each of the multi-recycled fuel cycles. The normalized values calculated for the MOX and UOX assemblies will be the input for a Perl script that will be used to generate a ORIGEN input deck. ORIGEN will the

be used to determine the ingestion toxicity of the two fuel types. The radiotoxicity of each fuel cycle will be compared against each other as well as the toxicity of the natural Uranium used in each of the fuel cycles.

B. DRAGON

To analyze a each fuel assembly and recycling scenario the fuel burnup, neutron distribution, and eigenvalue, k_{inf}, had to be determined. This is accomplished using the lattice physics code DRAGON.[19] The DRAGON code is a lattice physics computer code based on a modular design that was developed by the Institute of Nuclear Engineering in the Department of Mechanical Engineering at École Polytechnique de Montréal. The modules included in DRAGON allows the code to: calculate resonance self-shielding; perform isotopic depletion calculations; solve the multigroup neutron transport equation using either the collision probability method or the method of characteristics; perform multigroup probability integration; and a module to analyze numerous geometries and generate a tracking file for collision probability evaluation. DRAGON contains multiple cross section libraries such as JEFF2.2 and ENDF/B-VI for multiple energy groups. This is allows DRAGON to accurately analyze the isotopic evolution of the transuranics. The JEFF2.2 cross section library is used for the analyses carried out in this thesis.

C. ORIGEN

Once a recycling scenario has reached equilibrium or been recycled a significant number of times (15 cycles), the radiotoxicity of the waste is to be determined. DRAGON has the capabilities to simulate the decay of a large number of isotopes and is used in the two-year fabrication and five-year cooling periods prior to and after irradiation.

This code is unable to compute the radiotoxicity of a given isotopic composition using the dose equivalent factors. For this reason, ORIGEN[20] is used to carry out the radiotoxicity analysis. ORIGEN is a depletion and decay calculation sequence in the SCALE code package developed by the Oak Ridge National Laboratory. The ORIGEN solver is able to create a isotopic matrix and solve the production-destruction equations using an exponential matrix solver. With this solver ORIGEN can determine the isotopic concentration of a given fuel or waste over long periods of time. ORIGEN can also relate the isotopic concentration at a point in time to the radiological inhalation or ingestion hazards as well as the amount of decay heat being generated.

CHAPTER V

RESULTS AND DISCUSSION

Each of the fuel types described in Chapter III were evaluated following the procedure described in Chapter IV using the tools described in that chapter as well. This was done initially for a pin with a diameter of 0.41266 cm (nominal PWR fuel pellet) and for a smaller fuel pin to increase moderation. The results will be presented in terms of required ^{235}U enrichment, energy equivalence, voiding analysis, isotopic composition, transuranics (TRU) consumption, and ingestion radiotoxicity for each fuel type and both nominal pellet diameter and reduced pellet diameter.

A. Nominal Fuel Pellet Diameter

The required ^{235}U enrichment for a multi-recycled transuranic with Cm fuel assembly to achieve energy equivalence is presented in Table V-I. The required enrichment is initially above the 5 w/o limit and increases with each cycle. Because the ^{235}U enrichment is significantly greater than 5 w/o after the third cycle the study was ceased here and an alternate solution was examined. To reduce the enrichment additional moderation was sought. To gain the additional moderation a study was performed to examine the effects of a reduced pellet radius and various pellet shapes. The results of this study have been presented in Section III-G.

B. Reduced Fuel Pellet Diameter

Once it was determined that additional moderation was required to ensure the enrichment of ^{235}U remained below the 5 w/o, the recycling process was repeated for a fuel pellet with a reduced diameter for both multi-recycled fuel types.

Table V-I. Required ^{235}U enrichment to achieve energy equivalence for a recycle strategy with Cm for a standard pin radius.

Cycle	^{235}U Enrichment (wt%)
1	5.05
2	5.60
3	5.80

1. Required Uranium Enrichment

In both fuel types, the amount or weight percent (wt%) of TRU incluced in the fuel is constant at 8 wt%. So as to ensure energy equivalence, the enrichment of ^{235}U was altered for each cycle. Table V-II and Figure V-1 compare the enrichment required to achieve energy equivalence at each recycle for both the recycling strategy where Curium is included and the strategy where Curium is not recycled.

Table V-II. Required ^{235}U enrichment to achieve energy equivalence for each recycle strategy for a reduced pin radius.

Cycle	With Cm	Without Cm
1	3.00	3.02
2	4.35	4.52
3	4.73	5.06
4	4.82	5.29
5	4.83	5.40
6	4.83	5.47
7	4.82	5.52
8	4.82	5.55
9	4.81	5.58
10	4.81	5.60
11	4.81	5.61
12	4.80	5.62
13	4.80	5.63
14	4.80	5.64
15	4.79	5.64

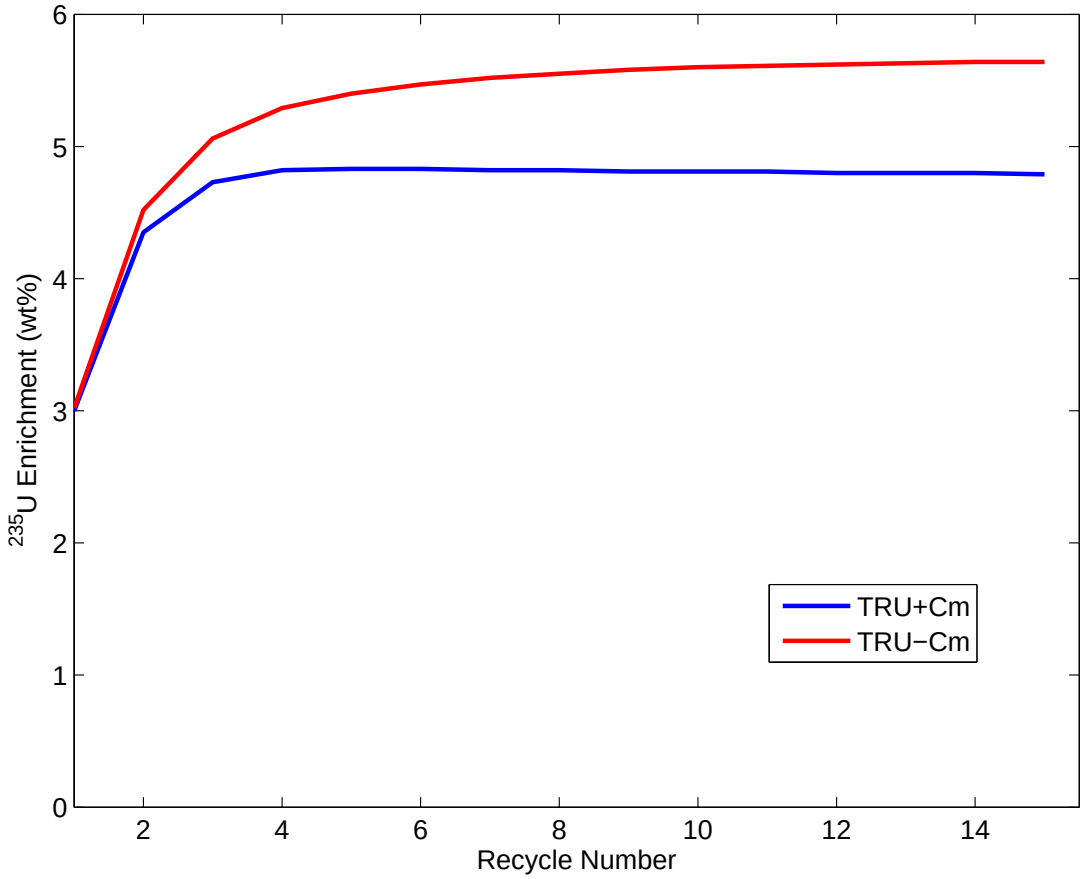


Fig. V-1. Required ^{235}U enrichment as a function of cycle for both recycling scenarios.

As can be seen in Fig. V-1, the necessary enrichment of ^{235}U from one cycle to the next initially increases for both cycles. This is the result of the degradation of the Plutonium vector where the amount of fissile Plutonium isotopes decreases with each recycle. The cause of the loss of the fissile isotopes is either a result of the fissile Pu isotopes fissioning, or transmuting to a heavier isotope resulting in a decrease in the reactivity of the fuel from one cycle to the next. As a result of these negative reactivity sources, additional fissile isotopes (^{235}U) have to be added to the fuel. Eventually, the sources of negative reactivity begin to reach an equilibrium slowing

the increase in the amount of ^{235}U added. It should be noted that the cycle where Curium is not recycled, the required enrichment exceeds the 5 wt% limit, resulting in this cycle considered to be unacceptable. Although the limit was exceeded, the study was continued on this fuel type for completeness.

The required enrichment for the fuel type with recycled Curium requires less ^{235}U than the cycle where Curium is left in the waste stream. This is the result of large fission cross section of ^{243}Cm, ^{244}Cm, ^{245}Cm, and ^{247}Cm. Figure V-2[21] presents the cross section of the fissile Curium isotopes. The addition of these isotopes results in additional fissions and subsequently an increase in the reactivity of the fuel. This results in the need for less ^{235}U to achieve energy equivalence with a standard UOX fuel. After the 6th cycle the additional reactivity resulting from the increase in fissile Cm isotopes causes the necessary ^{235}U enrichment to begin to decrease. Eventually the required ^{235}U enrichment will reach an equilibrium once fissile Cm isotopes reach an equilibrium.

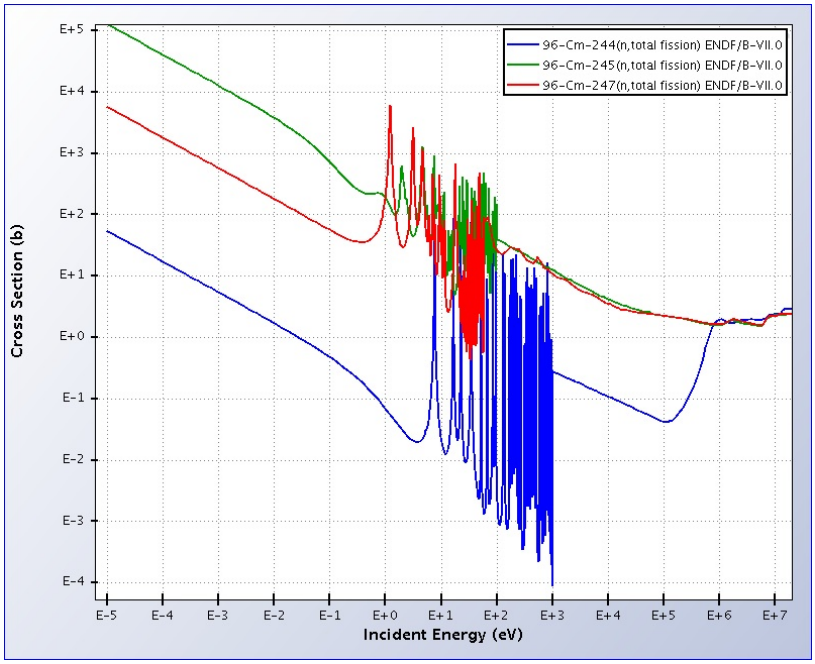


Fig. V-2. Fission cross section for select Curium isotopes.[21]

2. Isotopic Composition at the Beginning of Each Cycle

The following sections summarizes the TRU composition at the beginning of each cycle (BOC) for both recycling strategies. The values presented are for the mass (kg) of each isotope present in the reduced diameter fuel assembly. All of the values in the plots are tabulated and are included in Appendix C.

a. Neptunium Isotopic Evolution

The mass of Neptunium at the beginning of each cycle is plotted in Figure V-3. The plot represents the mass of ^{237}Np as this is the only long-lived Np isotope that is produced. ^{236}Np is not produced in significant quantities by this fuel. The other isotopes produced have almost completely decayed away after the completion of the cooling period (^{238}Np $t_{1/2}$ = 2.1 days, ^{239}Np $t_{1/2}$ = 2.4 days).

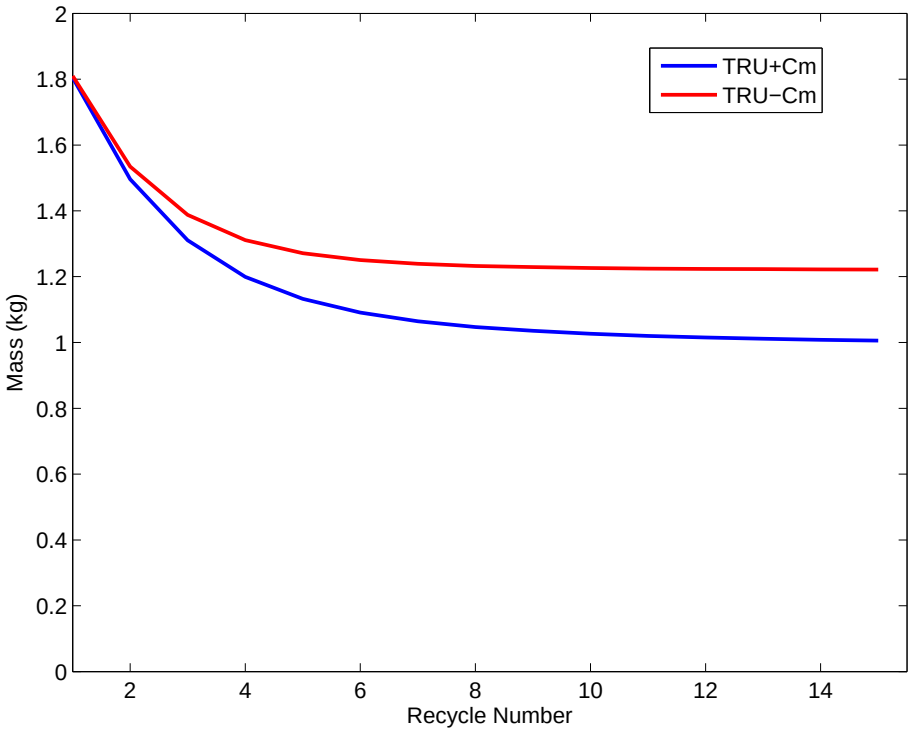


Fig. V-3. Evolution of ^{237}Np for 15 recycles.

^{237}Np is primarily produced in two processes. One way is through neutron capture in ^{235}U that leads to ^{236}U which then also absorbs a neutron, yielding ^{237}U. ^{237}U has a short half life ($t_{1/2}$ = 6.75 days) and β^- decays to ^{237}Np. The second manner is the alpha decay (α) of ^{241}Am ($t_{1/2}$ = 432 years). The legacy fuel used to create the fuel used in the first recycle has been cooled for 20 years allowing for the buildup of ^{237}Np as the Neptunium isotope has a much longer half life than ^{241}Am. As a result concentration of ^{237}Np increases during the cooling period. When the Neptunium is placed back in the reactor it is transmuted at a much faster rate than it is produced from the absorption of a neutron by ^{236}U or the decay of ^{241}Am. This along with the shorter cooling time allowing for a smaller buildup from the ^{241}Am decay causes the concentration to decrease initially before reaching and equilibrium.

b. Plutonium Isotopic Evolution

The mass of the five Plutonium isotopes found in the fuel at the beginning of the cycle are plotted in Figures V-4 and V-5. In the plots the values are in terms of kilograms in the assembly that was modeled.

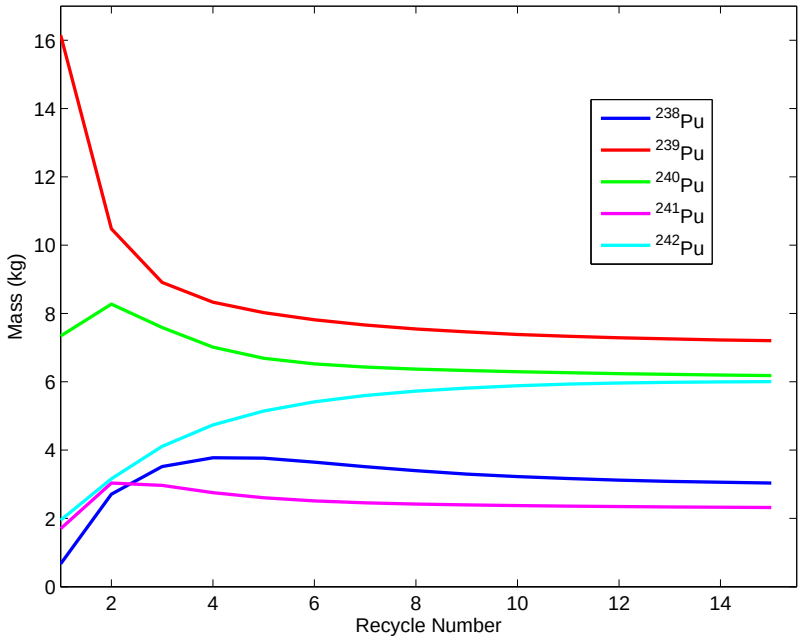


Fig. V-4. Evolution of Plutonium isotopes for the TRU+Cm recycling strategy.

The concentration of the isotopes presented above begin to stabilize after about 4 cycles except for ^{242}Pu which requires a few more cycles. The concentration of ^{238}Pu initially begins to increase before stabilizing. The cause of this increase is the transmutation of ^{237}Np, which absorbs a neutron and then beta decays (β^-) to ^{238}Pu. As was described in the previous section, the concentration of ^{237}Np increased during the 20 year cooling period. This caused a large concentration of ^{237}Np available to be transmuted to ^{238}Pu in the first cycle. As the concentration of ^{237}Np begin to decrease, the rate at which ^{238}Pu is produced also begins to decrease. Eventually the ^{238}Pu concentration reaches an equilibrium as the ^{237}Np concentration stabilizes.

The quantity of ^{239}Pu is largest prior to the first cycle then due to fission or transmutation the amount of ^{241}Pu begins to decrease. ^{239}Pu is produced through the transmutation of ^{238}U. The Uranium isotope absorbs a neutron becoming ^{239}U

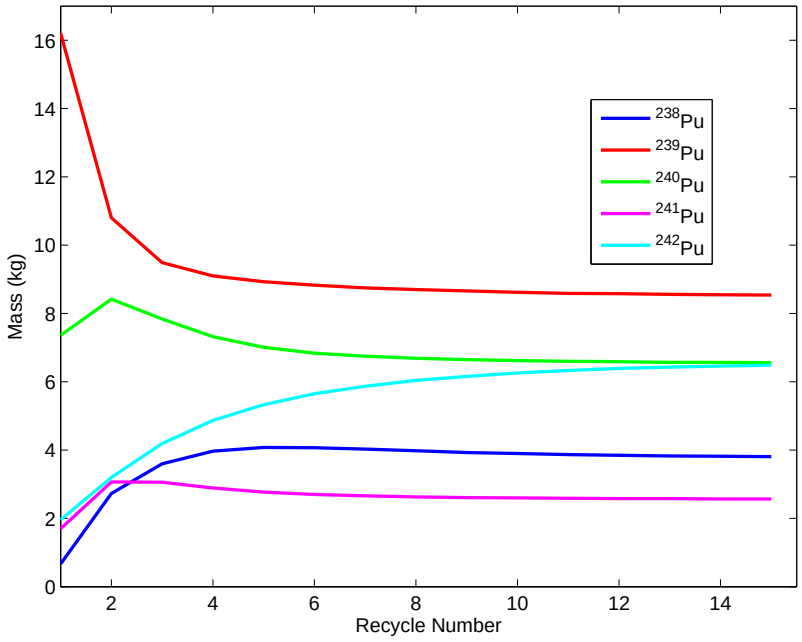


Fig. V-5. Evolution of Plutonium isotopes for the TRU-Cm recycling strategy.

($t_{1/2}$=23.45 min.) before β^- decaying to ^{239}Np which subsequently decays to ^{239}Pu. The ^{239}Pu is lost due to fission or transmutation quicker than it is produced and with the increased quantity of ^{235}U at each new cycle there is less ^{238}U to produce ^{239}Pu. Initially ^{240}Pu increases as the quantity of ^{239}Pu is the greatest but the quantity of ^{240}Pu begins to decrease as the amount produced decreases with the decrease in ^{239}Pu nuclides and the isotope is transmuted to ^{241}Pu. The ^{241}Pu follows the same behavior as ^{240}Pu while ^{242}Pu increases for the 15 cycles. This is because absorption cross section of ^{242}Pu is smaller than the lighter Plutonium isotopes resulting in a buildup of this isotope. As can be seen in Figures V-4 and V-5 above all five of the isotopes are beginning to approach an equilibrium as the rates of production and destruction reach a balance.

Figure V-6 compares the total amount of Plutonium present in the fuel assembly

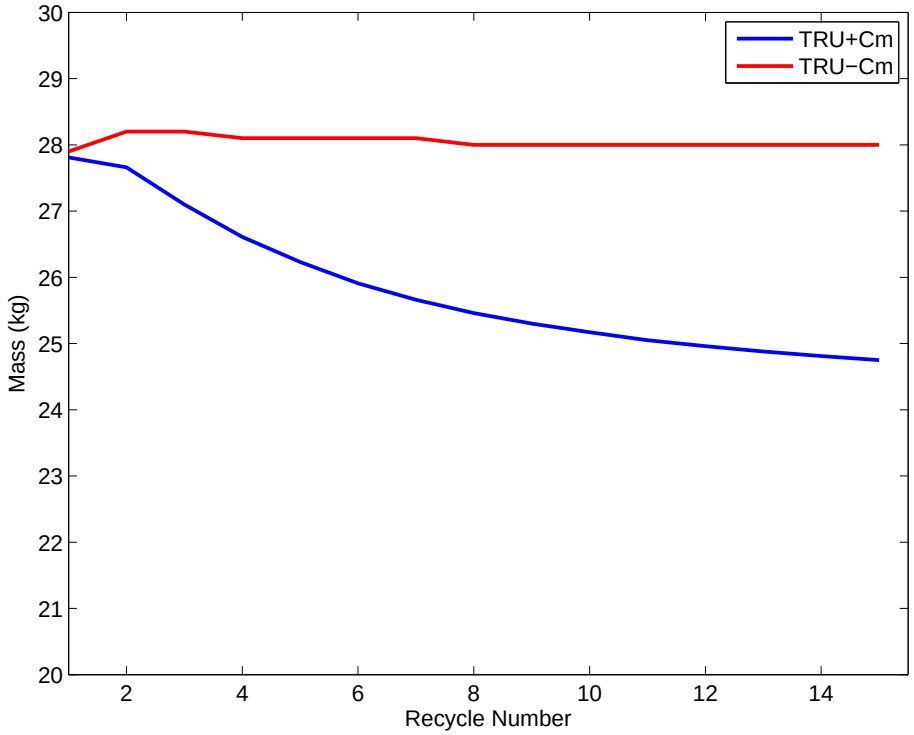


Fig. V-6. Total Plutonium mass at the beginning of each cycle for both fuels.

for both recycling strategies. The mass of Plutonium remains relatively constant for the fuel where Curium is not recycled while, in the other scenario, the mass of Plutonium begins to decreases slightly. This is the result of the buildup of the Curium isotopes. Figure V-7 shows the total mass of each element at the beginning of each cycle. The total mass of the transuranics remains constant although the mass of the Curium isotopes is increasing. The mass of the Pu, Np, and Am isotopes decreases to account for the increased Cm mass resulting in the difference seen in Figure V-6.

Figure V-8 is a comparison of the fissile Plutonium wt% of the Plutonium vector for the two recycling scenarios. In both cases, the fissile content decreases from about 65% before the first cycle to around 40% at the final cycle.

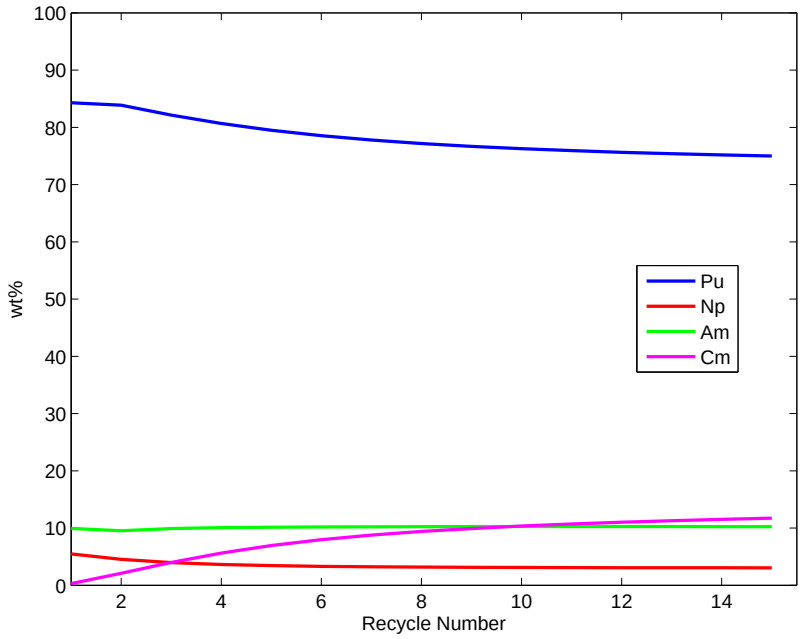


Fig. V-7. Mass of each element at the beginning of each cycle.

c. Americium Isotopic Evolution

The mass of the three Americium isotopes (^{241}Am, ^{242m}Am, and ^{243}Am) are plotted in Figures V-9 and V-10.

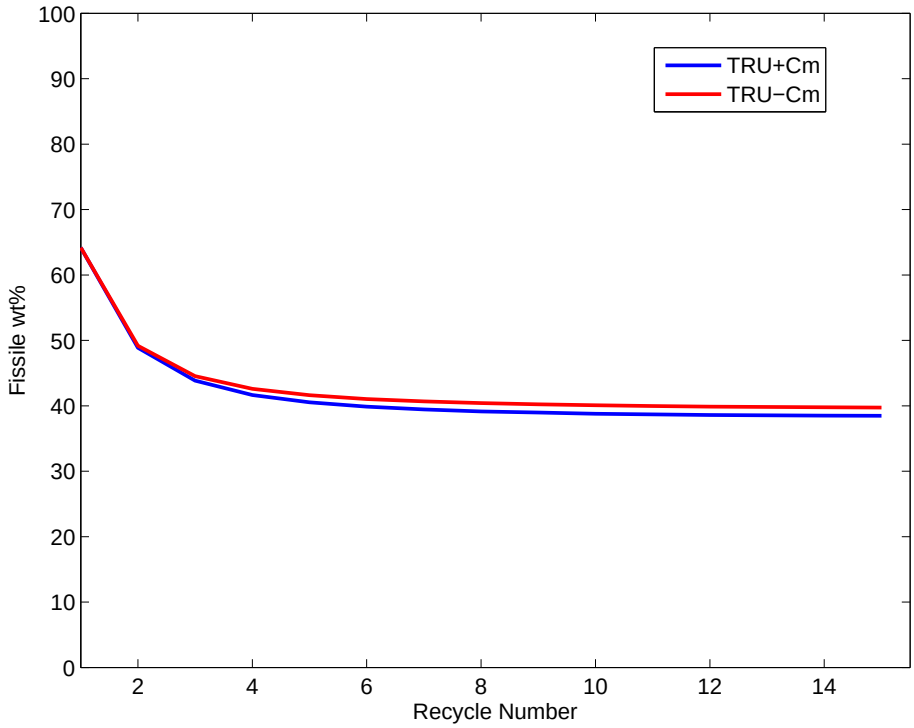


Fig. V-8. Fissile wt% of the Plutonium vector for the two recycling scenarios.

The mass of ^{241}Am is largest before the initial cycle but in the following cycles the destruction rate due to transmutation is greater than the production rate from ^{241}Pu. The reason the ^{241}Am isotope is largest initially is the result of the 20-year decay period before the legacy UOX was used to create new fuel. The ^{241}Pu isotope has a half-life of $t_{1/2}$=14.3 years. After the 20-year decay, over half of the ^{241}Pu has β^- decayed to ^{241}Am which has a much larger half-life ($t_{1/2}$=432 years) causing a buildup of the ^{241}Am isotope. Once placed in the reactor this isotope is transmuted to ^{242m}Am due to its large absorption cross section. After the first cycle there is only a 5 year cooling period allowing for less ^{241}Pu to decay into ^{241}Am resulting in a smaller quantity at the beginning of the second recycle. This trend continues until the production and destruction rates begin to reach an equilibrium.

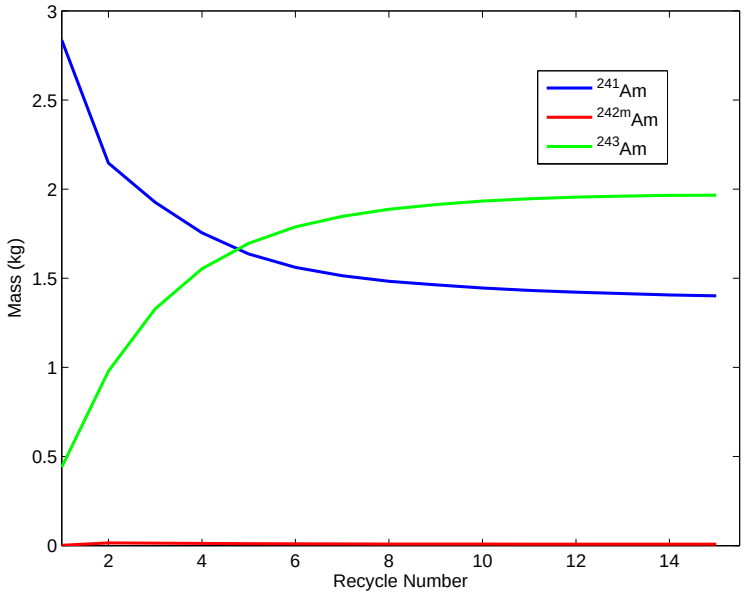


Fig. V-9. Mass of the Americium isotopes for the fuel cycle in which Curium is recycled.

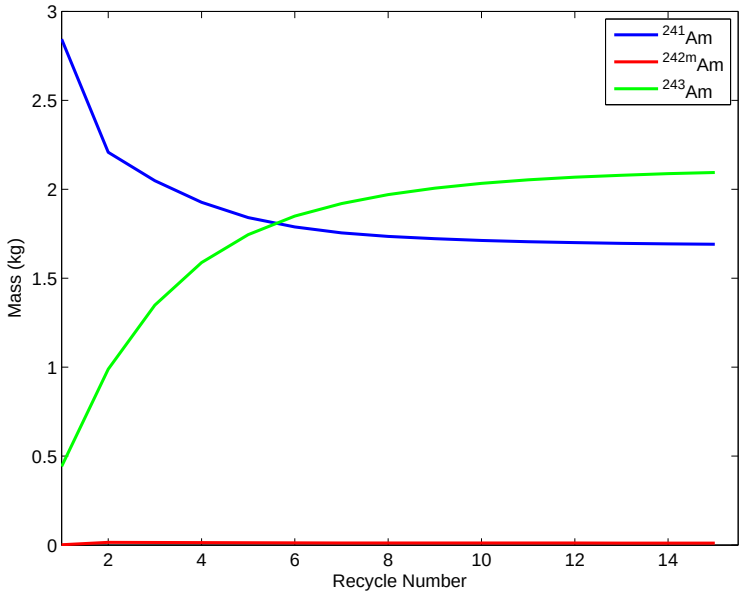


Fig. V-10. Mass of the Americium isotopes for the fuel cycle in which Curium is not recycled.

The quantity of ^{242m}Am ($t_{1/2}$ = 141 years) never reaches a significant quantity as the production rate from the transmutation of ^{241}Am is not much greater than the absorption rate. As a result the ^{241}Am isotope quickly transmutes to ^{243}Am. The absorption cross section of ^{243}Am is smaller than that of the other two Americium isotopes and ^{243}Am has long half-life ($t_{1/2}$=7370 years) resulting in a buildup of this isotope as seen in the figures above. The ^{243}Am production and destruction rates reach an equilibrium after the same occurs for ^{241}Am.

d. Curium Isotopic Evolution

The mass of significant Curium isotopes at the beginning of each recycle is presented in Figure V-11

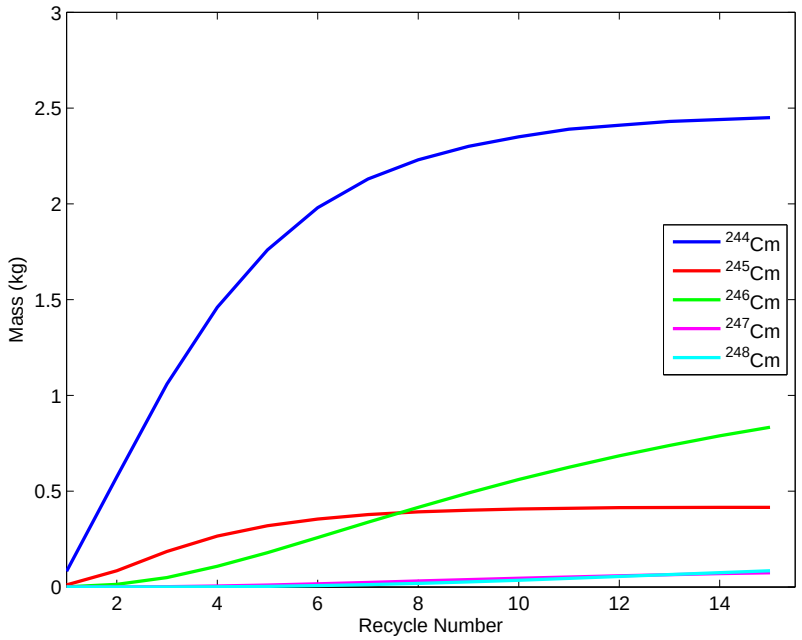


Fig. V-11. Mass of select Curium isotopes at the beginning of each recycle.

Curium is produced in a nuclear reactor from the transmutation of ^{243}Am. The ^{243}Am isotope absorbs a neutron becoming ^{244}Am which then β^- decays becoming ^{244}Cm. As a result, ^{243}Am is considered the gateway to heavier actinides. As the quantity of ^{243}Am increases so does the amount of ^{244}Cm which then translates into an increase in the amount of heavier Cm isotopes. ^{244}Cm is removed either through fission or more likely by absorbing a neutron producing ^{245}Cm. The fission cross section of ^{245}Cm is extremely large as seen in Figure V-2 and this isotope also has a large (n,γ) cross section. These two cross sections result in a large destruction rate which explains why the quantity of ^{245}Cm remains small. The portion of ^{245}Cm that is transmuted becomes ^{246}Cm which has a absorption cross section much smaller than that of ^{245}Cm. As a result, the amount of ^{246}Cm will buildup but will not exceed that of ^{244}Cm due to the fissioning of ^{245}Cm (these nuclides will not be able to transmute to ^{246}Cm). Since the production chain pass through ^{245}Cm before a nuclide becomes ^{246}Cm, the number of cycles needed to reach equilibrium for this isotope will lag behind that of ^{244}Cm significantly. After 15 cycles this isotope is still increasing at a significant rate. The next isotope in this chain is ^{247}Cm and, as was the case for ^{245}Cm, the quantity of this isotope does not increase to significant levels due to its absorption rate. ^{247}Cm transmutes to ^{248}Cm for which, for the same reasons as ^{246}Cm, the number of cycles to reach equilibrium significantly lag behind the other isotopes discussed. After 15 recycles, ^{248}Cm is just beginning to become noticeable and it may take many more recycles before an equilibrium is achieved.

3. Energy Equivalence

As described in Section III-C, the concept of energy equivalence was developed to ensure that the same amount of energy generated during any given cycle is identical between a multi-recycled assembly and a standard PWR UOX fuel assembly. Figures

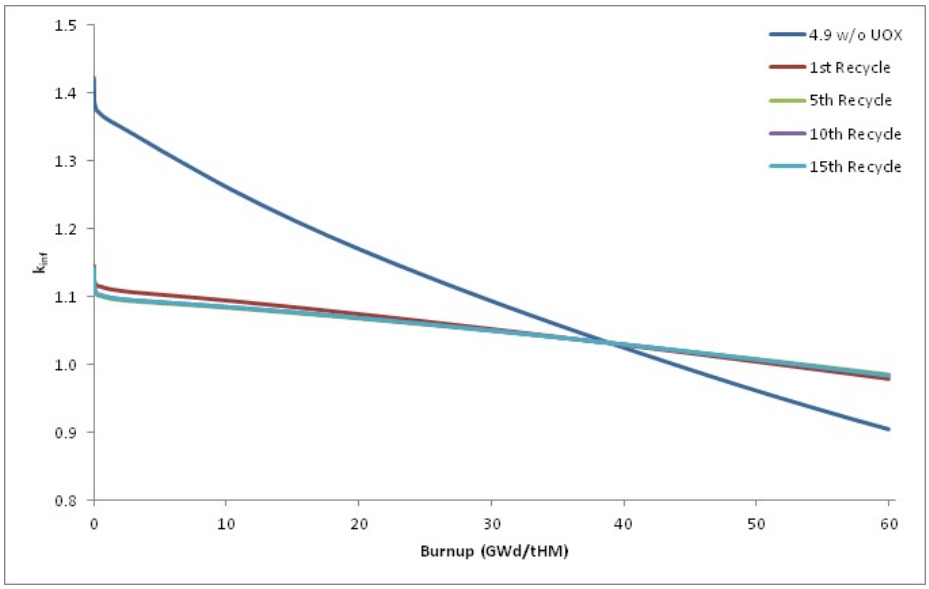


Fig. V-12. k_∞ as a function of burnup for standard UOX and TRU with Cm fuel assemblies.

V-12 and V-13 compare the values for k_∞ as a function of burnup for the recycled fuel assemblies and a standard UOX fuel assembly. Only the values for the 1^{st}, 5^{th}, 10^{th}, and 15^{th} recycle were plotted since there was not a large change in the values of k_∞ between the 1^{st} and 15^{th} recycle.

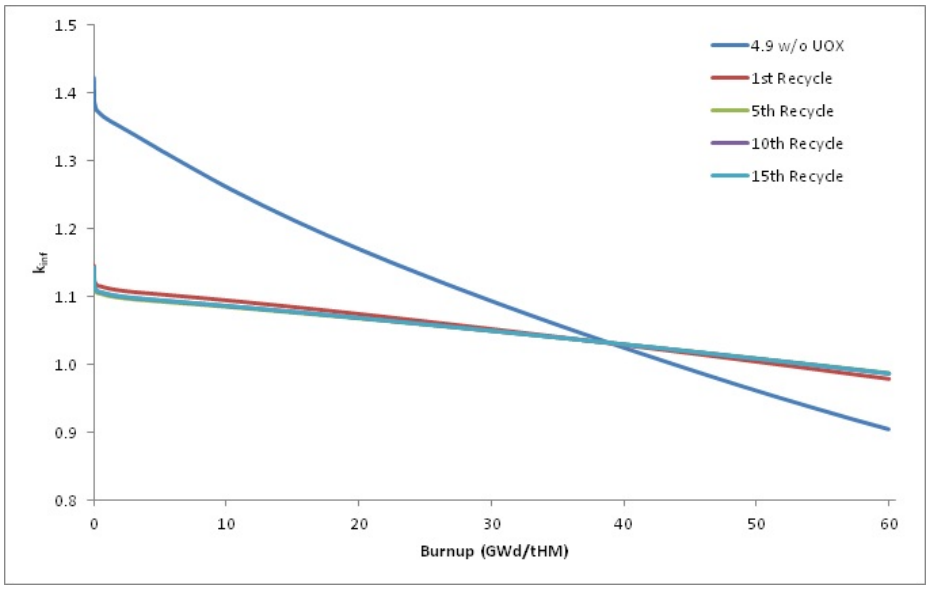


Fig. V-13. k_∞ as a function of burnup for standard UOX and TRU without Cm fuel assemblies.

The value of k_∞ plotted in Figures V-12 and V-13 all have a value of 1.035 at a burnup of 37.5 GWd/tHM and pass the energy equivalence criteria. The initial value of k_∞ is much greater for the UOX fuel assembly than the recycled fuel assemblies. The value is lower in the recycled assemblies because of the hardened neutron spectrum which is the result of the inclusion of the transuranics. There is very little change between the initial k_∞ value and the final k_∞ values for the recycled fuels while this is not the case for the UOX assembly. This is the result of the conversion of the fertile nuclides in the fuel (i.e., ^{238}Pu, ^{240}Pu) to fissile nuclides. There is a greater number of fertile isotopes in the recycled fuel than the UOX fuel since these isotopes are included initially resulting in a greater conversion rate to fissile isotopes.

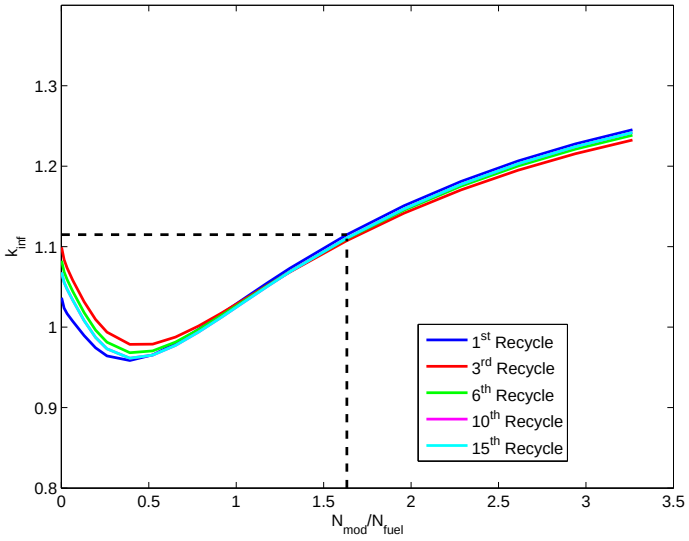


Fig. V-14. k_∞ as a function of the moderator to fuel ratio for the some of the recycled fuel assemblies where Cm is recycled.

4. Void Coefficient

The second criterion that was used to determine a successful fuel assembly is ensuring that k_∞ in an under-moderated fuel assembly does not exceed k_∞ for a standard fuel assembly. Figures V-14 and V-15 are a plot of k_∞ for various moderator to fuel ratios (N_{mod}/N_{fuel}). Only select recycles were used in the plot due to the fact that after the first cycle the values of k_∞ does not change drastically. The nominal moderator to fuel ratio for the reduced pin is denoted by the vertical black line.

The response of k_∞ to reduced moderation is very similar for the multi-recycled and MOX fuel assemblies. This was expected since a majority of the fuel is Plutonium which, as described in Chapter III, can cause an increase in reactivity when an assembly becomes under moderated. For the fuel type where Cm is recycled, the value of k_∞ always remains lower for reduced a reduced moderator to fuel ratio than for a standard ratio. This is not the case for the fuel type without Cm, only the first

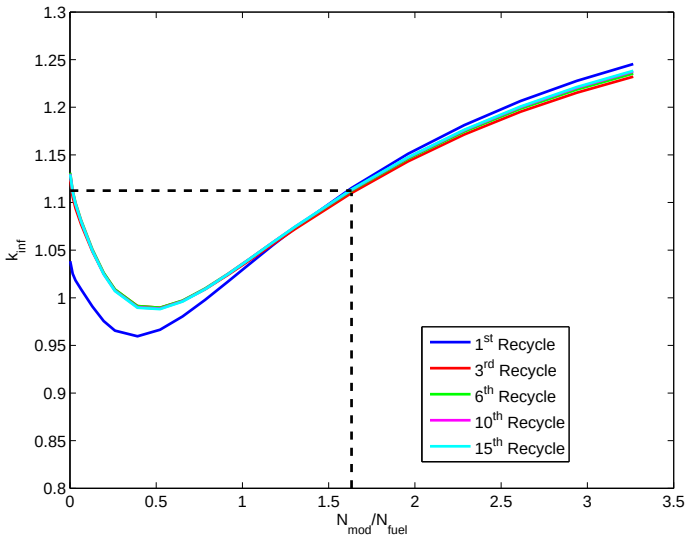


Fig. V-15. k_∞ as a function of the moderator to fuel ratio for the some of the recycled fuel assemblies where Cm is not recycled.

and second cycle has a negative moderator coefficient with the rest being positive. Always having a negative moderator coefficient was a requirement to be a successful fuel type, resulting in the fuel which Cm is not recycled failing a second criteria (also exceed the 5 w/o ^{235}U criteria). To ensure that the TRU-Cm fuel does not have a positive void coefficient, the weight percent of the transuranics in the fuel should be reduced.

The max value of k_∞ when the fuel assembly is fully voided (k_∞^{void}) for the fuel type where Cm is not recycled initially increases rapidly before stabilizing after the 3rd cycle and reaching a max value around the 6th cycle. The initial increase of k_∞^{void} is the result of production of ^{240}Pu and ^{242}Pu which results in a positive increase in the moderator coefficient. After the second recycle the amount of ^{240}Pu begins to decrease but the decreases is compensated by the increase of ^{242}Pu. Once the concentration of these two isotopes stabilizes, the value of k_∞^{void} stabilizes as well.

For the fuel type where Cm is recycled the value of k_{∞}^{void} also increases initially before reaching a max value during the 3rd recycle. After this recycle, the value of k_{∞}^{void} begins to decrease with each successive cycle. Once again the initial increase is the result of the production of ^{240}Pu and ^{242}Pu after the first few cycles. After the 3rd recycle, the value of k_{∞}^{void} begins to decrease with each cycle. It was shown in Figure V-6 that the amount of Plutonium contained in the fuel at the beginning of each cycle decreases for the fuel where Curium is recycled while it remains mostly constant when Curium is not recycled. As a result of the decreasing Plutonium concentration the value of k_{∞}^{void} decreases with each cycle. The inclusion of the fissile isotopes ^{245}Cm and ^{247}Cm also contributes to a more negative void coefficient. As the amount of moderation decreases the probability of ^{245}Cm and ^{247}Cm fissioning decreases resulting in a decrease of reactivity. Table V-III is a comparison of the reactivity change during the voiding of a fuel assembly. As expected the UOX fuel assembly had the greatest change between nominal conditions and voided. This is because almost all of the fissions that occur for a nominally moderated assembly are the result of ^{235}U but when moderation is lost the fission probability of ^{235}U decreases significantly. As mentioned in Chapter III, ^{238}U does fission in a fast neutron spectrum, but the absorption compensates for this increase in reactivity. The first MOX assembly referenced in the table has 8 wt% Pu and 2.1 wt% ^{235}U, the second assembly has 12 wt% Pu and 0.25 wt% ^{235}U.

5. Transuranics Production

An objective of multi-recycling is to reduce the amount of transuranics currently present in the legacy stockpile. To compare the effectiveness each fuel cycle the transuranics production rate was determined. Transuranics production rate is defined as the difference between the discharge (after 5 years of cooling) and the charge

Table V-III. Change in reactivity when moderation is lost.

Fuel Type	Recycle Number	$\Delta\rho$ (pcm)
UOX	—	-65842.3
MOX 8 wt%	—	-7974.7
MOX 12 wt%	—	8281.5
With Curium	1	-7842.8
	3	-841.4
	6	-2750.6
	15	-4273.6
Without Cm	1	-7688.8
	3	1353.1
	6	1861.0
	15	1585.5

(prior to the 2-year transportation). This difference is then normalized to 1 TWhe of electricity generated. The efficiency for converting from thermal to electrical power is 33%. For the legacy UOX and standard UOX fuel assemblies this process is straightforward, the difference between the input and output mass for each transuranics element divided by the amount of electricity generated by the fuel assembly. The process is not the same for MOX and the two multi-recycled assembly designs. Since the transuranics from one cycle are reused in the next, the charge mass for each cycle is the mass of the transuranics from the legacy UOX after 20 years of cooling multiplied by the number of legacy assemblies required to create the next recycle generation. The summation over the number of cycles is the transuranics charge for

a multi-recycled fuel. The discharge is the summation of the losses from recycling as well as the transuranics remaining following the final recycle. The number of legacy assemblies used in each fuel cycle is presented in Table V-IV. Tables V-V through V-IX presents the charge, discharge, and net production for each of the transuranic elements for every fuel type. A detailed mass balance for each recycle for the multi-recycled fuel assemblies are presented in Appendices A & B.

Table V-IV. Number of legacy assemblies required to create a recycled fuel assembly.

Fuel cycle	Number of Assemblies
MOX OTC	7.57
TRU with Cm (15 cycle)	23.26
TRU without Cm (15 cycles)	23.98

Table V-V. Charge, discharge and production rate for a legacy UOX assembly.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	0.00E+000	5.92E-001	5.92E-001
Pu239	0.00E+000	1.29E+001	1.29E+001
Pu240	0.00E+000	5.96E+000	5.96E+000
Pu241	0.00E+000	3.26E+000	3.26E+000
Pu242	0.00E+000	1.77E+000	1.77E+000
Np237	0.00E+000	1.34E+000	1.34E+000
Am241	0.00E+000	6.24E-001	6.24E-001
Am242m	0.00E+000	1.71E-003	1.71E-003
Am243	0.00E+000	3.91E-001	3.91E-001
Cm244	0.00E+000	1.36E-001	1.36E-001
Cm245	0.00E+000	9.23E-003	9.23E-003
Cm246	0.00E+000	1.10E-003	1.10E-003
Cm247	0.00E+000	1.31E-005	1.31E-005
Cm248	0.00E+000	8.78E-007	8.78E-007
Pu	0.00E+000	2.45E+001	2.45E+001
Np	0.00E+000	1.34E+000	1.34E+000
Am	0.00E+000	1.02E+000	1.02E+000
Cm	0.00E+000	1.47E-001	1.47E-001
Total TRU	0.00E+000	2.70E+001	2.70E+001

Table V-VI. Charge, discharge and production rate for a standard UOX assembly.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	0.00E+000	1.05E+000	1.05E+000
Pu239	0.00E+000	1.36E+001	1.36E+001
Pu240	0.00E+000	6.49E+000	6.49E+000
Pu241	0.00E+000	3.40E+000	3.40E+000
Pu242	0.00E+000	2.35E+000	2.35E+000
Np237	0.00E+000	2.00E+000	2.00E+000
Am241	0.00E+000	1.08E+000	1.08E+000
Am242m	0.00E+000	2.43E-003	2.43E-003
Am243	0.00E+000	6.15E-001	6.15E-001
Cm244	0.00E+000	2.55E-001	2.55E-001
Cm245	0.00E+000	2.27E-002	2.27E-002
Cm246	0.00E+000	3.31E-003	3.31E-003
Cm247	0.00E+000	5.08E-005	5.08E-005
Cm248	0.00E+000	4.35E-006	4.35E-006
Pu	0.00E+000	2.68E+001	2.68E+001
Np	0.00E+000	2.00E+000	2.00E+000
Am	0.00E+000	1.70E+000	1.70E+000
Cm	0.00E+000	2.81E-001	2.81E-001
Total TRU	0.00E+000	3.08E+001	3.08E+001

Table V-VII. Charge, discharge and production rate for a MOX assembly.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	4.05E+000	4.27E+000	2.22E-001
Pu239	9.78E+001	4.82E+001	-4.96E+001
Pu240	4.44E+001	3.82E+001	-6.28E+000
Pu241	1.03E+001	1.64E+001	6.11E+000
Pu242	1.18E+001	1.27E+001	9.00E-001
Np237	0.00E+000	1.01E+000	1.01E+000
Am241	0.00E+000	6.47E+000	6.47E+000
Am242m	0.00E+000	4.64E-002	4.64E-002
Am243	0.00E+000	3.81E+000	3.81E+000
Cm244	0.00E+000	2.15E+000	2.15E+000
Cm245	0.00E+000	3.66E-001	3.66E-001
Cm246	0.00E+000	3.04E-002	3.04E-002
Cm247	0.00E+000	6.59E-004	6.59E-004
Cm248	0.00E+000	4.34E-005	4.34E-005
Pu	1.68E+002	1.20E+002	-4.86E+001
Np	0.00E+000	1.01E+000	1.01E+000
Am	0.00E+000	1.03E+001	1.03E+001
Cm	0.00E+000	2.54E+000	2.54E+000
Total TRU	1.68E+002	1.34E+002	-3.48E+001

Table V-VIII. Charge, discharge and production rate for a TRU+Cm.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	8.30E-001	1.06E+000	2.28E-001
Pu239	2.00E+001	1.66E+000	-1.84E+001
Pu240	9.11E+000	1.77E+000	-7.34E+000
Pu241	2.11E+000	7.23E-001	-1.39E+000
Pu242	2.42E+000	1.94E+000	-4.81E-001
Np237	2.24E+000	2.55E-001	-1.98E+000
Am241	3.52E+000	3.43E-001	-3.17E+000
Am242m	2.44E-003	2.85E-003	4.14E-004
Am243	5.47E-001	6.47E-001	1.00E-001
Cm244	1.02E-001	8.17E-001	7.15E-001
Cm245	1.40E-002	1.41E-001	1.27E-001
Cm246	1.56E-003	2.15E-001	2.13E-001
Cm247	1.92E-005	1.80E-002	1.80E-002
Cm248	1.30E-006	1.54E-002	1.54E-002
Pu	3.45E+001	7.15E+000	-2.74E+001
Np	2.24E+000	2.55E-001	-1.98E+000
Am	4.07E+000	9.93E-001	-3.08E+000
Cm	1.18E-001	1.21E+000	1.09E+000
Total TRU	4.09E+001	9.61E+000	-3.13E+001

Table V-IX. Charge, discharge and production rate for a TRU-Cm.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	8.56E-001	1.27E+000	4.15E-001
Pu239	2.07E+001	1.88E+000	-1.88E+001
Pu240	9.39E+000	1.78E+000	-7.61E+000
Pu241	2.17E+000	7.75E-001	-1.40E+000
Pu242	2.50E+000	2.11E+000	-3.84E-001
Np237	2.31E+000	3.02E-001	-2.01E+000
Am241	3.63E+000	3.95E-001	-3.23E+000
Am242m	2.51E-003	3.77E-003	1.26E-003
Am243	5.64E-001	6.95E-001	1.30E-001
Cm244	0.00E+000	5.96E+000	5.96E+000
Cm245	0.00E+000	8.35E-001	8.35E-001
Cm246	0.00E+000	1.03E-001	1.03E-001
Cm247	0.00E+000	2.00E-003	2.00E-003
Cm248	0.00E+000	1.44E-004	1.44E-004
Pu	3.56E+001	7.81E+000	-2.78E+001
Np	2.31E+000	3.02E-001	-2.01E+000
Am	4.20E+000	1.09E+000	-3.10E+000
Cm	0.00E+000	6.91E+000	6.91E+000
Total TRU	4.21E+001	1.61E+001	-2.60E+001

The discharge values for the legacy UOX fuel assemblies is after 5 years of cooling and not the 20 years cooling that was used as the feed for the multi-recycling fuel. The legacy UOX, UOX 4.9, and MOX fuel assemblies have a standard fuel pellet radius of 0.41226 cm. The two TRU multi-recycled assemblies employ the reduced radius of 0.3557 cm.

The UOX 4.9 assembly is the greatest producer of Plutonium followed closely by the legacy UOX assemblies. This is as expected, since as discussed previously, Uranium transmutes to Plutonium. More Plutonium is created in the UOX 4.9 assembly than the legacy UOX due to its extended irradiation. The MOX OTC fuel cycle destroys the greatest amount of Plutonium but the TRU assemblies are more efficient destroying 79.27% for the fuel where Cm is recycled and 78.05% for the other multi-recycled fuel. The TRU assemblies transmute Pu more efficiently than the MOX OTC assembly. This is a result of the longer exposure of Plutonium in the reactor, yielding a greater probability that Plutonium will fission or transmute out of the Plutonium vector. The UOX 4.9 fuel assembly is also the greatest producer of Neptunium which is the result of containing a large amount of ^{235}U. The multi-recycled assembly where Curium is not recycled requires more ^{235}U than the standard UOX assembly, but since the Neptunium produced returns to the reactor multiple times, the Np that is produced from ^{235}U has a greater opportunity of being transmuted. The cycle where Curium is recycled destroys slightly less Np than the other multi-recycled fuel since some of the Np is displaced by the inclusion of Cm.

As expected the MOX OTC fuel created the greatest amount of Americium. Americium is produced by decay of ^{241}Pu or the absorption of a neutron by ^{242}Pu.When Pu is included in the fuel, the amount of ^{241}Pu produced during irradiation increases resulting in an increase in the production of ^{241}Am (^{241}Pu β decays to ^{241}Am). Since the MOX fuel includes the greatest amount of Pu initially, it follows that this fuel

type would be the greatest producer of Am as well. The multi-recycled assemblies are both burners of Am. Although they too contain Pu initially, the multiple irradiation cycle provides a greater opportunity for the Am produced from the Pu to be transmuted or fissioned.

None of the five fuel types examined were net burners of Curium with the legacy UOX assembly producing the least as a result of the shortest irradiation time. A shorter irradiation time results in fewer opportunities for an isotope to transmute to heavier actinides such as Cm. The multi-recycled fuel where Curium is not recycled generated the greatest amount of Curium. Since the Cm was sent to waste instead of being recycled, the amount of Cm increases steadily after each cycle while when Cm is recycled the amount of Cm in the fuel cycle will begin to reach an equilibrium as the production and destruction rates become equal.

Table V-X. Total TRU produced as well as the destruction ratio of the five fuel types.

Total TRU	Net Production (kg/TWhe)	Destruction Ratio
Legacy UOX	2.70E+01	0.00
UOX 4.9	3.08E+01	0.00
MOX OTC	-3.48E+01	1.25
TRU w/ Cm	-3.13E+01	4.26
TRU w/o Cm	-2.60E+01	2.61

Table V-X presents the total transuranic produced by each cycle as well as the destruction ratio. The destruction ratio is the amount of transuranics charged to a fuel type divided by the amount of transuranics discharged. The amount discharged is

once again the summation of the waste produced after each recycle and the transuranics in the fuel after the 15^{th} recycle. Although the MOX OTC fuel has the greatest destruction rate, the destruction ratio is the lowest amongst the three recycled fuel types. The multi-recycle assemblies have a greater destruction ratio than the MOX OTC since the transuranics that were not transmuted after one cycle are placed back into the reactor increasing the likelihood that they will be transmuted. The two UOX fuels have a destruction ratio of 0 since these fuel do not destroy the transuranics only produce them.

A second method to examine the transuranics produced is to consider the comprehensive transuranics production. The transuranics comprehensive production is the amount of waste produced by a fuel assembly including the waste as a result of the fabrication of the fuel assembly. The total waste is then normalized based on the total electricity generated. For a UOX assembly the comprehensive transuranics production is just the mass of transuranics remaining after irradiation divided by the power produced by the fuel assembly. For the MOX and multi-recycled fuel assemblies the comprehensive transuranics production is the transuranics that become waste as well as the transuranics from the legacy UOX assembly that were not used to create the recycled assembly. For example, the comprehensive production for a MOX fuel assembly would be the mass of transuranics after irradiation (since none are recycled again) plus the mass of Np, Am, Cm and 0.1% Pu from the legacy assemblies used to create the MOX fuel. This combined mass would then be normalized by the power produced from the MOX and legacy assemblies. Figure V-16 compares the amount of waste produced by each cycle. The benifits of recycling Pu once can be seen by the difference between the UOX and MOX lines. By muliti-recycling the transuranics, the amount of TRU waste generated decreases significantly with the greatest gains the result of recycling Pu, Np, Am, and Cm. Following the 15^{th} cycle the fuel is

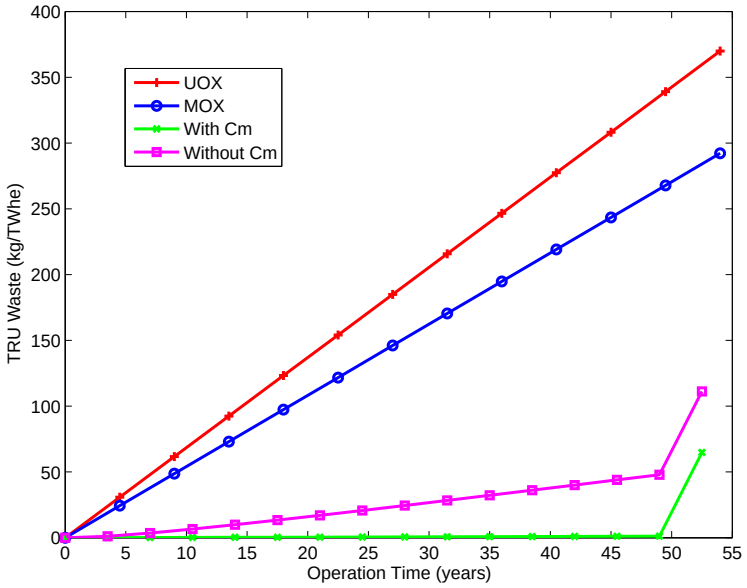


Fig. V-16. Amount of TRU produced by each fuel type.

discharged resulting in the jump in the amount of waste in the multi-recycled fuels. Even after all of the fuel is discharged, the multi-recycled fuels have generated a smaller amount of transuranics than the direct and once through cycles.

6. Radiotoxicity

The radiotoxicity results were obtained using the ORIGEN code as described in Chapter IV. All of the radiotoxicity values are for ingestion toxicity per TWhe. The radiotoxicity of each fuel type is normalized to the toxicity of the natural Uranium required to create the fuel. Since each fuel type requires different amounts of Unat, the toxicity of Unat will be different for each fuel. Table V-XI is the ingestion radiotoxicity of Unat for each of the fuel types. The ingestion toxicity has been normalized to the power produced by the fuel type.

Figure V-17 is a comparison of the radiotoxicity of the four fuel types (UOX,

Table V-XI. Ingestion toxicity of the natural Uranium required for each fuel type.

Fuel Type	Ingestion Toxicity
UOX	8.99E+08
MOX	8.47E+08
TRU with Cm	8.77E+08
TRU without Cm	9.19E+08

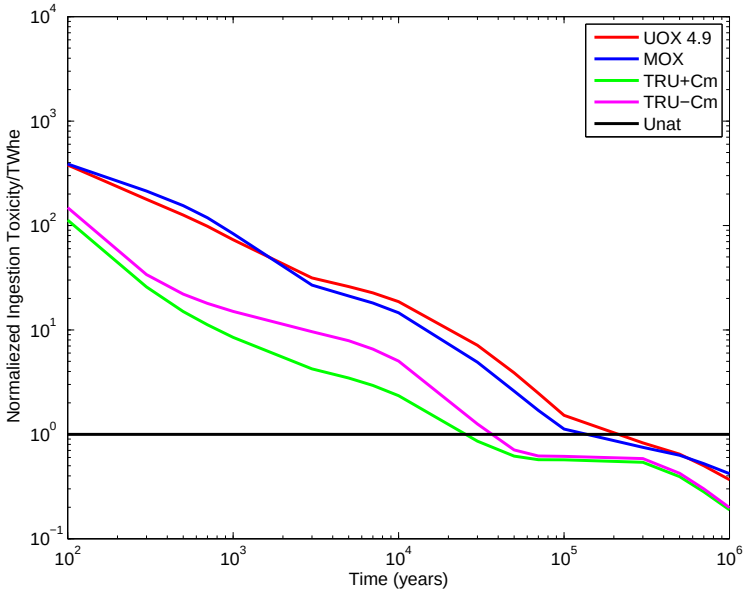


Fig. V-17. Radiotoxicity of each fuel type normalized to the electricity generated and the toxicity of the natural Uranium.

MOX and multi-recycle with and without Cm) normalized to the toxicity of the natural Uranium required to create the fuel assemblies. The two multi-recycled assemblies were each recycled 15 times with the waste from the first 14 cycles being the fission products, actinides that were not recycled and the losses from the recycle process. After the 15^{th} recycle all of fuel was sent to the waste storage with none of the transuranics recycled. The standard pin radius was used for the UOX and MOX fuel assemblies and the multi-recycle assemblies had a reduced radius. Figure V-17 demonstrates the benefit of multi-recycling as the radiotoxicity of the two multi-recycled assemblies is significantly lower than the MOX and UOX assemblies. The time required for the ingestion toxicity to fall below the ingestion toxicity of Unat is shown in Table V-XII. By multi-recycling the transuranics, the time required can be reduced by about a factor of 9 if Cm is recycled or a factor of 4.5 when Cm is not recycled.

Table V-XII. Amount of time required for the radiotoxicity of each fuel type to fall below the radiotoxicity of Unat.

Fuel Type	Time (years)
UOX	249,700
MOX	162,600
With Cm	27,940
Without Cm	56,620

Figure V-18 and V-19 compare the radiotoxicity of the two fuel type for a select number of recycles. The change in the radiotoxicity of both fuels types initially decreases significantly as the number of recycles increases but as the composition of the fuel begins to reach an equilibrium, the change in radiotoxicity decreases.

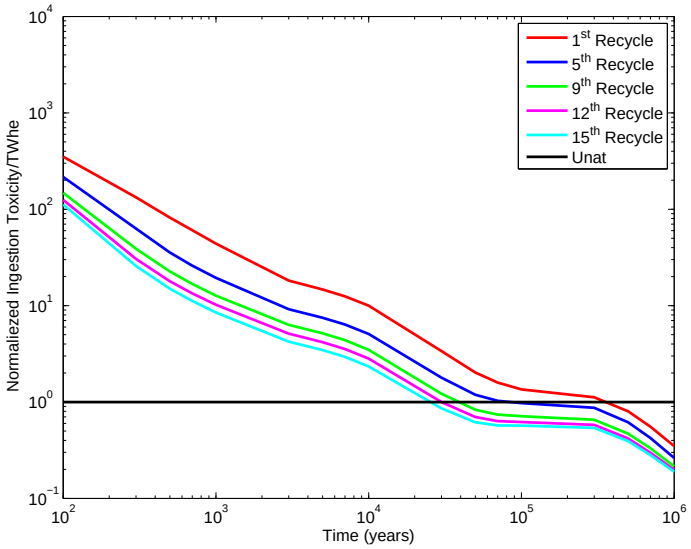


Fig. V-18. Radiotoxicity of TRU+Cm normalized to the electricity generated and the toxicity of the natural Uranium for select cycles.

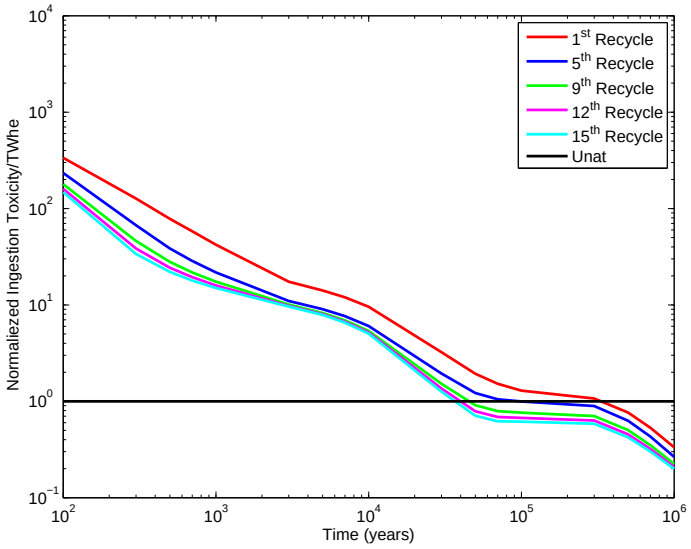


Fig. V-19. Radiotoxicity of TRU-Cm normalized to the electricity generated and the toxicity of the natural Uranium for select cycles.

Table V-XIII. Amount of time required for the radiotoxicity of each fuel type to fall below the radiotoxicity of Unat when the 15^{th} cycle is not discharged.

Fuel Type	Time (years)
UOX	249,700
MOX	162,600
With Cm	304.1
Without Cm	26,880

If it is assumed that after the 15^{th} cycle, the fuel is not completely discharged but instead the recycling process is continued, the the radiotoxicity of the waste after the 15^{th} would become lower. Figure V-20 compares the radiotoxicity of a UOX, MOX, and the two multi-recycled fuels when the recycling process continues. The radiotoxicity of values for the two multi-recycled assemblies is similar to what one would expect where an isotopic equilibrium reached since after 15 recycles the composition of the fuel is mostly stable from one cycle to the next. The amount of time required for the toxicity of the fuel to fall below the value of natural Uranium is shown in Table V-XIII. By keeping the transuranics in the fuel cycle it follows that the radiotoxicity of the fuel will decrease. The most significant gains by not discharging the 15^{th} cycle is obtained for the TRU+Cm fuel. The reason is as follows: if the fuel is discharge after 15 recycles a large amount of Curium that had been accumulation in the fuel will be added to the waste at this point. When only the waste from this 15^{th} cycle is added to the waste then only a small portion of this Curium (0.1%) leaves the fuel cycle with the bulk remaining in the fuel not contributing to the radiotoxicity of the waste.

It was shown in Section III - H that the radiotoxicity of the waste is initially a

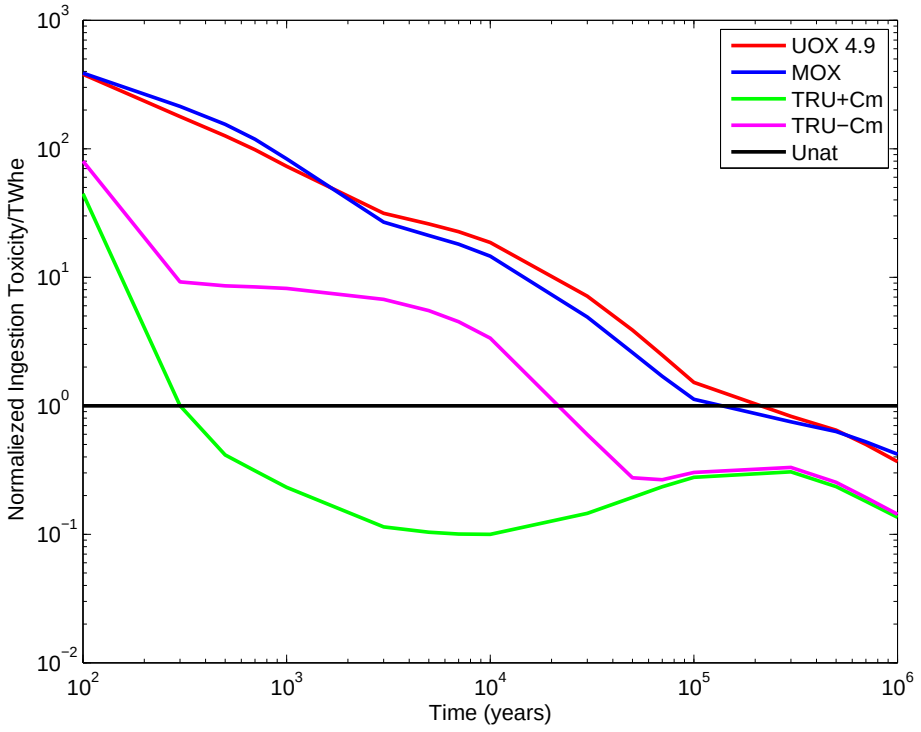


Fig. V-20. Radiotoxicity of each fuel type normalized to the electricity generated and the toxicity of the natural Uranium when the 15th is not discharged.

function of the radiotoxicity of the fission products and short-lived isotopes. After about one hundred years to some thousands of years the total radiotoxicity of the fuel is mostly driven by the toxicity of the actinides specifically for these fuel types considered, ^{239}Pu, ^{240}Pu, and for a short time ^{241}Am. These isotopes are of large enough quantity and have a significant enough dose factor that they are the most significant contributors to ingestion toxicity during this time period. Figures V-21 through V-23 compares of the toxicity of each of these isotopes. The radiotoxicity of each isotope is after 15 recycles (all of the fuel discharge after the 15th recycle) and has been normalized based on the electricity generated and the toxicity of the required amount natural Uranium.

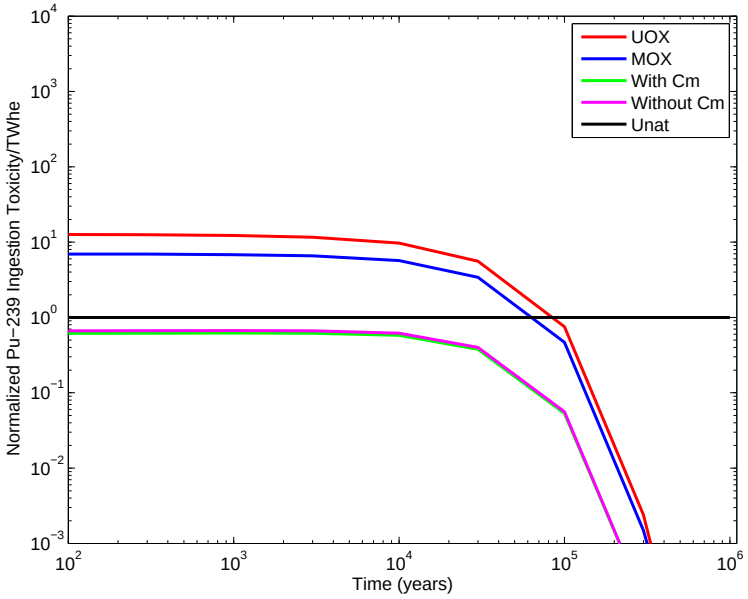


Fig. V-21. Radiotoxicity of ^{239}Pu for each fuel type normalized to the radiotoxicity of natural Uranium.

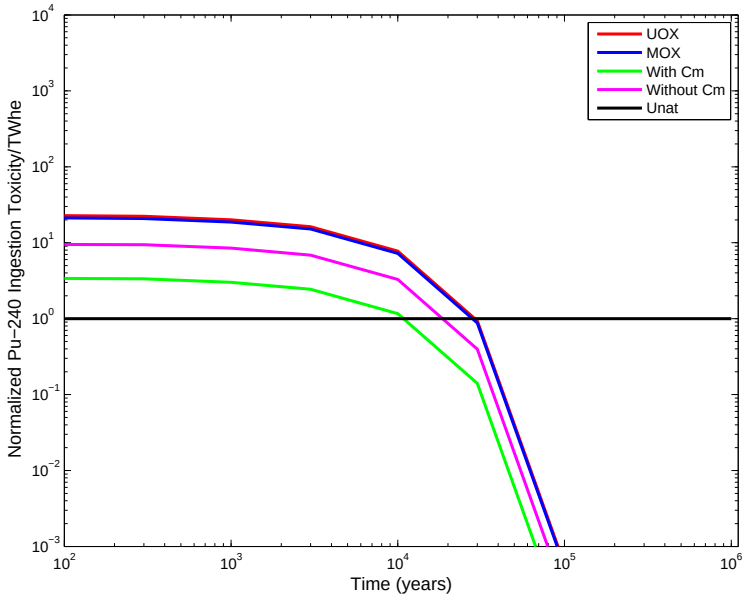


Fig. V-22. Radiotoxicity of ^{240}Pu for each fuel type normalized to the radiotoxicity of natural Uranium.

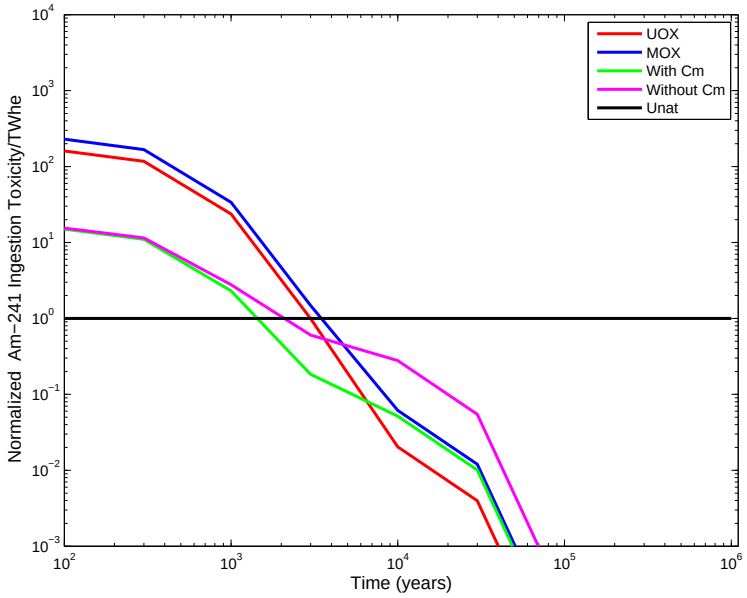


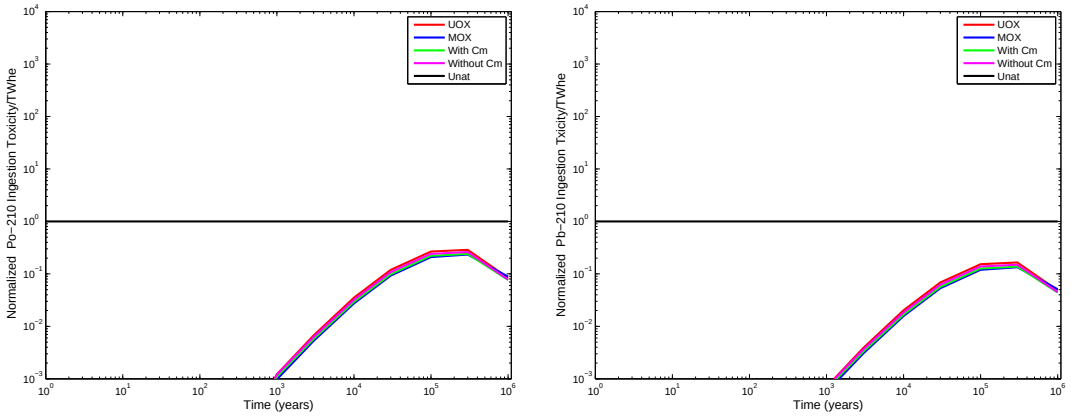
Fig. V-23. Radiotoxicity of ^{241}Am for each fuel type normalized to the radiotoxicity of natural Uranium.

The benefit of recycling Pu can be seen in Figures V-21 and V-22. The toxicity of the MOX and two multi-recycled fuels are both lower than the UOX fuel assembly since they both have reduced amount of Pu after recycling. There is only a slight difference between the toxicity of ^{239}Pu for the two multi-recycled fuel with the TRU+Cm have a lower radiotoxicity due to this fuel greater ^{239}Pu consumption rate (lower quantity in the waste). There is a noticeable difference between the radiotoxicity of ^{240}Pu for the two multi-recycled fuels which is the result of the amount of ^{244}Cm in the waste. ^{244}Cm α-decays to ^{240}Pu having a half life of 18.1 years. After 100 years a majority of ^{244}Cm has decayed to ^{240}Pu. The cycle where Curium is not recycled produces 8.3 times more ^{244}Cm than the other multi-recycled assembly. As a result, the radiotoxicity of ^{240}Pu is lower for the TRU+Cm fuel cycle. The radiotoxicity of ^{241}Am for the two multi-recycled fuel assemblies initially is about the same

but the radiotoxicity of the TRU+Cm begins to decrease at a faster rate than the other multi-recycled fuel. This is the result of the decay of ^{245}Cm. The TRU fuel that does not recycle Cm produces more ^{245}Cm than the other fuel types, as a result this isotope decays to ^{241}Pu which then decays to ^{241}Am. This additional production of ^{241}Am results in the radiotoxicity of the TRU fuel without Cm to decrease at a slower rate. The toxicity of this fuel cycle eventually exceeds that of the MOX fuel as a result of the ^{245}Cm decay. These three isotopes (^{239}Pu, ^{240}Pu, and ^{241}Am) are the main contributors to the radiotoxicity values from about 100 years to a few thousands years. Since for all three isotopes the TRU fuel where Cm is recycled the radiotoxicity values were the lowest of the four fuels it follows that it has the lowest radiotoxicity in this period. Due to the additional toxicity of ^{240}Pu and ^{241}Am for the fuel without Cm when compared to the TRU with Cm fuel, the overall toxicity during this time period will be higher for the fuel without Cm than the fuel with Cm. The effect of the additional ^{240}Pu and ^{241}Am from ^{245}Cm can be seen on the overall radiotoxicity plot for the TRU fuel without Cm where the slope of the line decreases from 1,000 to 10,000 years.

After about 100,000 years the radiotoxicity of the waste levels off for both of the multi-recycled fuels and less drastically for the UOX and MOX fuel. The cause of this is the accumulation of ^{210}Po, ^{210}Pb, and to a lesser extent ^{226}Ra and ^{230}Th. These four isotopes are near the end of the decay chain of ^{238}U and require a significant amount of time to be produced. Numerous isotopes feed into this decay chain including, ^{234}U, ^{242}Pu, ^{242}Am (electron capture branching ratio of 17%), and ^{246}Cm. Figure V-24 is a plot of the toxicity of ^{210}Po and ^{210}Pb respectively. In the figures it can clearly be seen how both isotopes begin to accumulate late in the decay time line. The difference in the radiotoxicity of each is not significant and because most of the remaining radiotoxicity is the result of these isotopes, the radiotoxicity of the four

fuel assembly to begin to converge.

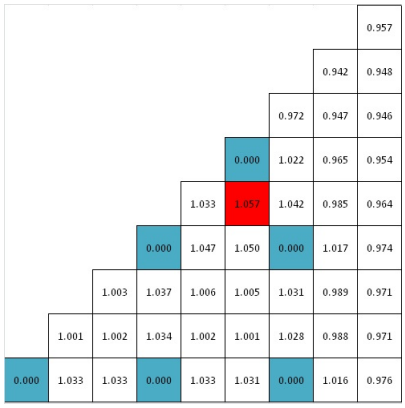


(a) Radiotoxicity of ^{210}Po (b) Radiotoxicity of ^{210}Pb

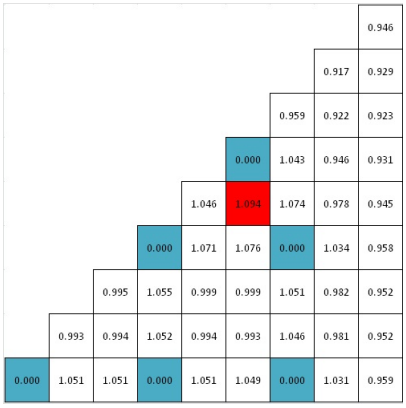
Fig. V-24. Radiotoxicity of ^{210}Po and ^{201}Pb for each fuel type normalized to the radiotoxicity of natural Uranium.

7. Pin Power Peaking Factors

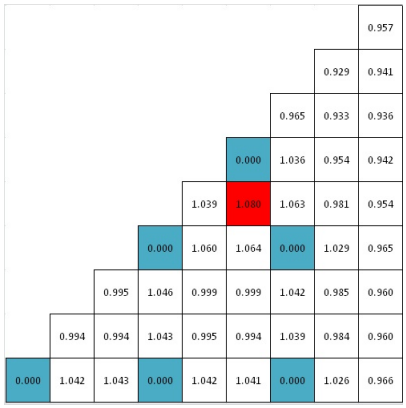
The final results that will be presented are the pin power peaking factors. These values provide a comparison of the amount of power generated in one pin as compared to the average pin power. Only the values for $1/8^{\text{th}}$ of a fuel assembly are necessary due to symmetry. None of the power peaking factors in the four fuel types exceed 1.1 with the largest pin power peaking factor found in the MOX fuel assembly. There is not a significant change between the 1^{st} and the 15^{th} cycle. Figure V-25 presents the values of the power peaking factor for each fuel type.



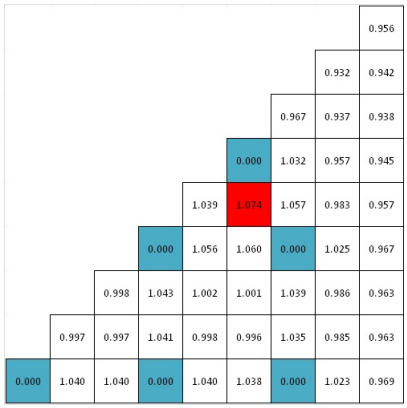
(a) UOX Power Peaking Factors



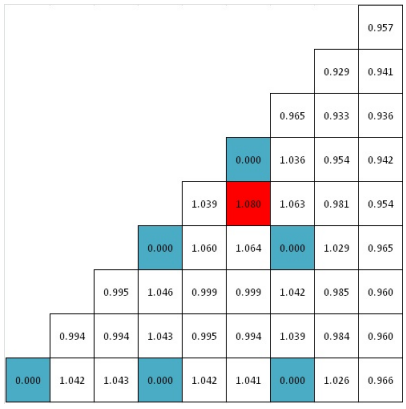
(b) MOX Power Peaking Factors



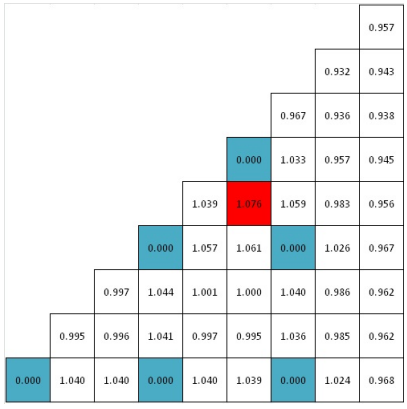
(c) TRU+Cm 1st recycle Power Peaking Factors



(d) TRU+Cm 15th recycle Power Peaking Factors



(e) TRU-Cm 1st recycle Power Peaking Factors



(f) TRU-Cm 15th recycle Power Peaking Factors

Fig. V-25. Power peaking factors at a burnup of 0 GWd/tHM.

CHAPTER VI

CONCLUSIONS AND RECOMMENDATIONS FOR FUTURE WORK

In this study, two multi-recycled fuel assemblies were examined and compared to a UOX and MOX once through cycle. The two multi-recycled fuel assemblies examined the possibilities of recycling Plutonium, Neptunium, Americium, and one examined recycling Curium as well. The four fuels were compared based on energy equivalence, void coefficient analysis, transuranic production, and radiotoxicity.

It was initially found that in order to multi-recycle the transuranics in a standard PWR fuel assembly additional moderation would be required. After performing a study analyzing the optimal pin shape or diameter it was found that a standard fuel pin with a radius of 0.3557 cm was the best option. This diameter reduced the required ^{235}U enrichment while only shortening the campaign length by 1 year. This diameter was used for both multi-recycled fuel assemblies in the study.

An objective of this study was to determine if isotopic equilibrium could be achieved in a PWR fuel assembly so as to keep the transuranics in the fuel cycle indefinitely. A isotopic equilibrium was not achieved after 15 cycles but it was noted that the composition of the fuel was only changing slightly from cycle to cycle indicating that and equilibrium could be achieved if the recycle process was to be continued.

For the energy equivalence study it was desired that every fuel assembly that the value for k_∞ at the average burnup (37.5 GWd/tHM) be equal to 1.035 to account for leakage. To achieve the ^{235}U enrichment was adjusted. A limit was set at 5 w/o ^{235}U, which a fuel assembly must remain below to be considered acceptable. The fuel assembly where Curium is recycled (TRU+Cm) had a maximum ^{235}U enrichment of 4.83 w/o and as the amount of Cm being recycled accumulated, the ^{235}U enrichment

began to decrease. For the fuel assembly where Cm was not recycled (TRU-Cm), the enrichment of ^{235}U exceeded 5 w/o after the second recycle and continued to increase through the 15th recycle. As a result the TRU-Cm assemblies are not considered acceptable. For completeness, the analysis was continued on this fuel type. A voiding analysis was performed on the two multi-recycled assemblies as well. The criteria set was that the value of k_∞ for any voided condition should not exceed the value of k_∞ for standard moderator conditions. Once again it was found that the TRU-Cm fuel assemblies violated this condition after the second cycle. The TRU+Cm fuel assembly was found to not violate this criteria for any cycle and the value of k_∞ for a fully voided moderator was found to decrease after reaching a maximum at the 3rd recycle.

When considering transuranic production, a fuel can be judged based on amount of transuranics produced/destroyed or the efficiency by which the transuranic is produced/destroyed. It was found that a MOX once through cycle (OTC) destroyed the largest quantity of transuranics at -34.8 kg/TWhe. The TRU+Cm fuel destroys a comparable amount of transuranics, eliminating 31.3 kg/TWhe. Both multi-recycled assemblies as well as the MOX fuel were net burners of transuranics while as expected the UOX assembly is a transuranics producer. The TRU+Cm was the most efficient transuranics burner, with an efficiency of 76.53%. The amount of transuranic elements each fuel type destroyed or produced was also examined. Once again the MOX fuel eliminated the greatest amount of Pu, -48.6 kg/TWhe, while this fuel assembly did produce Np (1.01 kg/TWhe), Am (10.3 kg/TWhe) and Cm (2.54 kg/TWhe). The TRU-Cm fuel destroyed less Pu than the MOX fuel (-27.8 kg/TWhe) but was a net burner of Np (-2.01 kg/TWhe) and Am (-3.10 kg/TWhe). The TRU-Cm fuel was the greatest producer of Cm creating 6.91 kg/TWhe. The TRU+Cm fuel was a net burner of Pu (-27.4 kg/TWhe), Np (-1.98 kg/TWhe) and Am (-3.08 kg TWhe), but

similar to the other fuel types, was generated Cm at a rate of 1.09 kg/TWhe. Based on these results it is seen that no significant benefit is gained from the TRU-Cm fuel that is not also gained by the TRU+Cm fuel. The TRU+Cm is actually more beneficial as it produces less Cm than the other multi-recycled fuel. As a result the TRU+Cm assembly is the best assembly for eliminating transuranics.

A radiotoxicity study was also performed in this study to determine if multi-recycling the transuranics provided any benefits. To compare each cycle, it was determined much time was required for the ingestion toxicity of each cycle to fall below the toxicity of the natural Uranium required to create the fuel. It was found that the time for the toxicity of UOX to fall below the toxicity of natural Uranium for was 2.49×10^5 years for the UOX 4.9 cycle and 1.63×10^5 years for MOX. If after the 15^{th} cycle all of the transuranics in the fuel cycle were sent to waste the the amount of time required for the ingestion toxicity of TRU recycling strategies to fall below natural Uranium is 5.66×10^4 years for TRU-Cm and 2.79×10^4 years the TRU+Cm. Once again multi-recycling Pu, Np, Am, and Cm was found to be the most beneficial. The amount of time is reduced even further if the waste is not discharged but instead the recycling process is assumed to continue. If so then it will require 2.69×10^4 years for TRU-Cm and only 304 years for TRU+Cm to fall below the toxicity of natural Uranium.

The multi-recycling strategy where Cm is recycled was found to be the most beneficial fuel type. This fuel type did not exceed the ^{235}U enrichment limit and did not violate the void coefficient criteria while the other multi-recycled assembly violated both limits. The TRU+Cm fuel assembly was also more efficient at eliminating transuranics and had a lower ingestion radiotoxicity than the other three fuels examined. With a reduced fuel pin diameter a TRU+Cm fuel could be multi-recycled.

Future work should focus on achieving an isotopic equilibrium. This was not

achieved in this study but the fuel composition was beginning to stabilize from one cycle to the next. A study should also consider reactor physics parameters such as fuel temperature coefficient of reactivity and control rod worth. A more in depth study of the effects of Am and Cm on the void coefficient of reactivity should also be performed. Finally, at the end of irradiation, an appreciable amount of ^{235}U remains, a study should be performed as to the benefits of extracting this isotope and recycling it as well.

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APPENDIX A

TRANSURANIC DESTRUCTION: TRU WITH CURIUM

The amount of transuranics charged and discharged as well as the production rate for each of the 15 recycles for the TRU+Cm recycling strategy. The discharged fuel in the following tables is the amount of transuranics remaining after irradiation and the five year cooling period. All of the values have been normalized to the amount of electricity generated by the fuel assembly.

Charge, discharge, and production rate for a TRU+Cm 1st recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	3.41E+000	1.30E+001	9.57E+000
Pu239	8.24E+001	3.28E+001	-4.96E+001
Pu240	3.75E+001	3.28E+001	-4.62E+000
Pu241	8.67E+000	1.33E+001	4.66E+000
Pu242	9.96E+000	1.36E+001	3.68E+000
Np237	9.21E+000	5.32E+000	-3.89E+000
Am241	1.45E+001	7.32E+000	-7.16E+000
Am242m	1.00E-002	7.50E-002	6.50E-002
Am243	2.25E+000	4.43E+000	2.19E+000
Cm244	4.20E-001	2.83E+000	2.41E+000
Cm245	5.78E-002	4.20E-001	3.63E-001
Cm246	6.42E-003	6.73E-002	6.09E-002
Cm247	7.91E-005	1.78E-003	1.70E-003
Cm248	5.33E-006	2.07E-004	2.01E-004
Pu	1.42E+002	1.06E+002	-3.64E+001
Np	9.21E+000	5.32E+000	-3.89E+000
Am	1.67E+001	1.18E+001	-4.91E+000
Cm	4.84E-001	3.32E+000	2.84E+000
Total TRU	1.68E+002	1.26E+002	-4.23E+001

Charge, discharge, and production rate for a TRU+Cm 2nd recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.38E+001	1.73E+001	3.46E+000
Pu239	5.35E+001	2.92E+001	-2.43E+001
Pu240	4.22E+001	3.14E+001	-1.09E+001
Pu241	1.55E+001	1.34E+001	-2.05E+000
Pu242	1.61E+001	1.90E+001	2.89E+000
Np237	7.63E+000	4.88E+000	-2.76E+000
Am241	1.09E+001	6.98E+000	-3.97E+000
Am242m	7.74E-002	6.83E-002	-9.09E-003
Am243	5.00E+000	6.34E+000	1.34E+000
Cm244	2.93E+000	5.35E+000	2.42E+000
Cm245	4.35E-001	9.38E-001	5.03E-001
Cm246	6.88E-002	2.52E-001	1.83E-001
Cm247	1.80E-003	9.73E-003	7.93E-003
Cm248	2.08E-004	1.71E-003	1.51E-003
Pu	1.41E+002	1.10E+002	-3.08E+001
Np	7.63E+000	4.88E+000	-2.76E+000
Am	1.60E+001	1.34E+001	-2.64E+000
Cm	3.44E+000	6.55E+000	3.12E+000
Total TRU	1.68E+002	1.35E+002	-3.31E+001

Charge, discharge, and production rate for a TRU+Cm 3rd recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.79E+001	1.87E+001	7.34E-001
Pu239	4.55E+001	2.79E+001	-1.76E+001
Pu240	3.87E+001	2.91E+001	-9.59E+000
Pu241	1.52E+001	1.25E+001	-2.61E+000
Pu242	2.10E+001	2.24E+001	1.46E+000
Np237	6.69E+000	4.49E+000	-2.20E+000
Am241	9.83E+000	6.39E+000	-3.44E+000
Am242m	7.03E-002	6.08E-002	-9.44E-003
Am243	6.78E+000	7.53E+000	7.57E-001
Cm244	5.43E+000	7.40E+000	1.97E+000
Cm245	9.49E-001	1.35E+000	3.98E-001
Cm246	2.53E-001	5.48E-001	2.95E-001
Cm247	9.74E-003	2.71E-002	1.74E-002
Cm248	1.71E-003	6.81E-003	5.09E-003
Pu	1.38E+002	1.11E+002	-2.76E+001
Np	6.69E+000	4.49E+000	-2.20E+000
Am	1.67E+001	1.40E+001	-2.69E+000
Cm	6.65E+000	9.33E+000	2.69E+000
Total TRU	1.68E+002	1.39E+002	-2.98E+001

Charge, discharge, and production rate for a TRU+Cm 4th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.93E+001	1.86E+001	-6.22E-001
Pu239	4.25E+001	2.70E+001	-1.55E+001
Pu240	3.58E+001	2.78E+001	-7.98E+000
Pu241	1.41E+001	1.18E+001	-2.24E+000
Pu242	2.42E+001	2.46E+001	4.03E-001
Np237	6.12E+000	4.22E+000	-1.90E+000
Am241	8.96E+000	5.90E+000	-3.06E+000
Am242m	6.25E-002	5.41E-002	-8.43E-003
Am243	7.93E+000	8.28E+000	3.56E-001
Cm244	7.47E+000	8.94E+000	1.47E+000
Cm245	1.36E+000	1.62E+000	2.69E-001
Cm246	5.49E-001	9.15E-001	3.66E-001
Cm247	2.71E-002	5.35E-002	2.63E-002
Cm248	6.80E-003	1.79E-002	1.10E-002
Pu	1.36E+002	1.10E+002	-2.60E+001
Np	6.12E+000	4.22E+000	-1.90E+000
Am	1.69E+001	1.42E+001	-2.71E+000
Cm	9.41E+000	1.16E+001	2.14E+000
Total TRU	1.68E+002	1.40E+002	-2.84E+001

Charge, discharge, and production rate for a TRU+Cm 5th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.92E+001	1.81E+001	-1.14E+000
Pu239	4.10E+001	2.63E+001	-1.47E+001
Pu240	3.41E+001	2.71E+001	-7.02E+000
Pu241	1.33E+001	1.14E+001	-1.89E+000
Pu242	2.62E+001	2.60E+001	-2.54E-001
Np237	5.78E+000	4.05E+000	-1.73E+000
Am241	8.35E+000	5.58E+000	-2.77E+000
Am242m	5.58E-002	4.95E-002	-6.22E-003
Am243	8.65E+000	8.76E+000	1.01E-001
Cm244	9.00E+000	1.00E+001	1.04E+000
Cm245	1.63E+000	1.80E+000	1.72E-001
Cm246	9.15E-001	1.32E+000	4.01E-001
Cm247	5.34E-002	8.62E-002	3.28E-002
Cm248	1.78E-002	3.63E-002	1.85E-002
Pu	1.34E+002	1.09E+002	-2.50E+001
Np	5.78E+000	4.05E+000	-1.73E+000
Am	1.71E+001	1.44E+001	-2.67E+000
Cm	1.16E+001	1.33E+001	1.66E+000
Total TRU	1.68E+002	1.41E+002	-2.77E+001

Charge, discharge, and production rate for a TRU+Cm 6th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.86E+001	1.74E+001	-1.23E+000
Pu239	3.99E+001	2.58E+001	-1.41E+001
Pu240	3.33E+001	2.68E+001	-6.53E+000
Pu241	1.28E+001	1.11E+001	-1.68E+000
Pu242	2.76E+001	2.70E+001	-6.57E-001
Np237	5.57E+000	3.94E+000	-1.63E+000
Am241	7.97E+000	5.38E+000	-2.58E+000
Am242m	5.12E-002	4.67E-002	-4.49E-003
Am243	9.12E+000	9.07E+000	-5.51E-002
Cm244	1.01E+001	1.08E+001	7.06E-001
Cm245	1.81E+000	1.92E+000	1.08E-001
Cm246	1.32E+000	1.73E+000	4.09E-001
Cm247	8.62E-002	1.23E-001	3.65E-002
Cm248	3.63E-002	6.26E-002	2.63E-002
Pu	1.32E+002	1.08E+002	-2.42E+001
Np	5.57E+000	3.94E+000	-1.63E+000
Am	1.71E+001	1.45E+001	-2.64E+000
Cm	1.34E+001	1.46E+001	1.28E+000
Total TRU	1.68E+002	1.41E+002	-2.72E+001

Charge, discharge, and production rate for a TRU+Cm 7th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.79E+001	1.68E+001	-1.15E+000
Pu239	3.91E+001	2.53E+001	-1.38E+001
Pu240	3.28E+001	2.65E+001	-6.28E+000
Pu241	1.25E+001	1.10E+001	-1.57E+000
Pu242	2.86E+001	2.76E+001	-9.11E-001
Np237	5.43E+000	3.87E+000	-1.56E+000
Am241	7.73E+000	5.26E+000	-2.47E+000
Am242m	4.83E-002	4.48E-002	-3.49E-003
Am243	9.43E+000	9.28E+000	-1.50E-001
Cm244	1.09E+001	1.13E+001	4.70E-001
Cm245	1.93E+000	1.99E+000	6.56E-002
Cm246	1.73E+000	2.13E+000	4.00E-001
Cm247	1.23E-001	1.61E-001	3.79E-002
Cm248	6.25E-002	9.61E-002	3.36E-002
Pu	1.31E+002	1.07E+002	-2.37E+001
Np	5.43E+000	3.87E+000	-1.56E+000
Am	1.72E+001	1.46E+001	-2.62E+000
Cm	1.47E+001	1.57E+001	1.01E+000
Total TRU	1.68E+002	1.41E+002	-2.69E+001

Charge, discharge, and production rate for a TRU+Cm 8th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.73E+001	1.63E+001	-1.02E+000
Pu239	3.85E+001	2.50E+001	-1.35E+001
Pu240	3.25E+001	2.64E+001	-6.13E+000
Pu241	1.24E+001	1.09E+001	-1.50E+000
Pu242	2.92E+001	2.81E+001	-1.09E+000
Np237	5.34E+000	3.83E+000	-1.52E+000
Am241	7.57E+000	5.17E+000	-2.40E+000
Am242m	4.64E-002	4.35E-002	-2.83E-003
Am243	9.63E+000	9.42E+000	-2.11E-001
Cm244	1.14E+001	1.17E+001	3.06E-001
Cm245	2.00E+000	2.04E+000	4.00E-002
Cm246	2.13E+000	2.51E+000	3.81E-001
Cm247	1.60E-001	1.98E-001	3.76E-002
Cm248	9.61E-002	1.36E-001	3.99E-002
Pu	1.30E+002	1.07E+002	-2.32E+001
Np	5.34E+000	3.83E+000	-1.52E+000
Am	1.72E+001	1.46E+001	-2.61E+000
Cm	1.58E+001	1.66E+001	8.04E-001
Total TRU	1.68E+002	1.42E+002	-2.66E+001

Charge, discharge, and production rate for a TRU+Cm 9th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.68E+001	1.59E+001	-9.02E-001
Pu239	3.81E+001	2.48E+001	-1.33E+001
Pu240	3.23E+001	2.63E+001	-6.04E+000
Pu241	1.22E+001	1.08E+001	-1.46E+000
Pu242	2.97E+001	2.85E+001	-1.20E+000
Np237	5.28E+000	3.79E+000	-1.49E+000
Am241	7.46E+000	5.11E+000	-2.36E+000
Am242m	4.51E-002	4.26E-002	-2.48E-003
Am243	9.76E+000	9.51E+000	-2.49E-001
Cm244	1.18E+001	1.19E+001	1.96E-001
Cm245	2.05E+000	2.07E+000	2.27E-002
Cm246	2.50E+000	2.86E+000	3.58E-001
Cm247	1.98E-001	2.34E-001	3.61E-002
Cm248	1.36E-001	1.80E-001	4.48E-002
Pu	1.29E+002	1.06E+002	-2.29E+001
Np	5.28E+000	3.79E+000	-1.49E+000
Am	1.73E+001	1.47E+001	-2.61E+000
Cm	1.66E+001	1.73E+001	6.57E-001
Total TRU	1.68E+002	1.42E+002	-2.64E+001

Charge, discharge, and production rate for a TRU+Cm 10th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.64E+001	1.56E+001	-8.16E-001
Pu239	3.77E+001	2.46E+001	-1.31E+001
Pu240	3.21E+001	2.61E+001	-5.98E+000
Pu241	1.21E+001	1.07E+001	-1.43E+000
Pu242	3.00E+001	2.87E+001	-1.29E+000
Np237	5.24E+000	3.77E+000	-1.47E+000
Am241	7.38E+000	5.05E+000	-2.32E+000
Am242m	4.41E-002	4.18E-002	-2.28E-003
Am243	9.86E+000	9.59E+000	-2.77E-001
Cm244	1.20E+001	1.21E+001	1.20E-001
Cm245	2.08E+000	2.09E+000	1.20E-002
Cm246	2.86E+000	3.19E+000	3.32E-001
Cm247	2.34E-001	2.68E-001	3.41E-002
Cm248	1.80E-001	2.29E-001	4.84E-002
Pu	1.28E+002	1.06E+002	-2.26E+001
Np	5.24E+000	3.77E+000	-1.47E+000
Am	1.73E+001	1.47E+001	-2.60E+000
Cm	1.74E+001	1.79E+001	5.46E-001
Total TRU	1.68E+002	1.42E+002	-2.62E+001

Charge, discharge, and production rate for a TRU+Cm 11th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.62E+001	1.54E+001	-7.47E-001
Pu239	3.74E+001	2.44E+001	-1.30E+001
Pu240	3.20E+001	2.60E+001	-5.93E+000
Pu241	1.20E+001	1.06E+001	-1.41E+000
Pu242	3.03E+001	2.89E+001	-1.36E+000
Np237	5.20E+000	3.75E+000	-1.45E+000
Am241	7.31E+000	5.01E+000	-2.30E+000
Am242m	4.34E-002	4.13E-002	-2.11E-003
Am243	9.93E+000	9.63E+000	-2.97E-001
Cm244	1.22E+001	1.23E+001	6.72E-002
Cm245	2.10E+000	2.10E+000	4.96E-003
Cm246	3.19E+000	3.50E+000	3.05E-001
Cm247	2.68E-001	2.99E-001	3.18E-002
Cm248	2.29E-001	2.79E-001	5.07E-002
Pu	1.28E+002	1.05E+002	-2.24E+001
Np	5.20E+000	3.75E+000	-1.45E+000
Am	1.73E+001	1.47E+001	-2.60E+000
Cm	1.80E+001	1.84E+001	4.60E-001
Total TRU	1.68E+002	1.42E+002	-2.60E+001

Charge, discharge, and production rate for a TRU+Cm 12th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.59E+001	1.52E+001	-7.01E-001
Pu239	3.72E+001	2.43E+001	-1.29E+001
Pu240	3.18E+001	2.59E+001	-5.89E+000
Pu241	1.20E+001	1.06E+001	-1.39E+000
Pu242	3.04E+001	2.90E+001	-1.41E+000
Np237	5.18E+000	3.74E+000	-1.44E+000
Am241	7.25E+000	4.97E+000	-2.28E+000
Am242m	4.28E-002	4.07E-002	-2.07E-003
Am243	9.98E+000	9.66E+000	-3.12E-001
Cm244	1.23E+001	1.24E+001	3.15E-002
Cm245	2.11E+000	2.11E+000	-9.10E-004
Cm246	3.50E+000	3.77E+000	2.80E-001
Cm247	2.99E-001	3.29E-001	2.93E-002
Cm248	2.79E-001	3.31E-001	5.20E-002
Pu	1.27E+002	1.05E+002	-2.23E+001
Np	5.18E+000	3.74E+000	-1.44E+000
Am	1.73E+001	1.47E+001	-2.59E+000
Cm	1.85E+001	1.89E+001	3.92E-001
Total TRU	1.68E+002	1.42E+002	-2.59E+001

Charge, discharge, and production rate for a TRU+Cm 13th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.57E+001	1.51E+001	-6.55E-001
Pu239	3.70E+001	2.42E+001	-1.28E+001
Pu240	3.17E+001	2.59E+001	-5.85E+000
Pu241	1.19E+001	1.05E+001	-1.37E+000
Pu242	3.05E+001	2.91E+001	-1.45E+000
Np237	5.16E+000	3.73E+000	-1.43E+000
Am241	7.21E+000	4.95E+000	-2.27E+000
Am242m	4.23E-002	4.04E-002	-1.87E-003
Am243	1.00E+001	9.68E+000	-3.20E-001
Cm244	1.24E+001	1.24E+001	4.64E-003
Cm245	2.12E+000	2.12E+000	-2.57E-003
Cm246	3.77E+000	4.03E+000	2.55E-001
Cm247	3.28E-001	3.55E-001	2.69E-002
Cm248	3.31E-001	3.83E-001	5.22E-002
Pu	1.27E+002	1.05E+002	-2.22E+001
Np	5.16E+000	3.73E+000	-1.43E+000
Am	1.73E+001	1.47E+001	-2.59E+000
Cm	1.89E+001	1.93E+001	3.36E-001
Total TRU	1.68E+002	1.42E+002	-2.59E+001

Charge, discharge, and production rate for a TRU+Cm 14th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.56E+001	1.50E+001	-6.27E-001
Pu239	3.69E+001	2.41E+001	-1.28E+001
Pu240	3.16E+001	2.58E+001	-5.83E+000
Pu241	1.19E+001	1.05E+001	-1.37E+000
Pu242	3.06E+001	2.91E+001	-1.47E+000
Np237	5.14E+000	3.72E+000	-1.43E+000
Am241	7.17E+000	4.92E+000	-2.25E+000
Am242m	4.19E-002	4.01E-002	-1.84E-003
Am243	1.00E+001	9.70E+000	-3.28E-001
Cm244	1.25E+001	1.24E+001	-1.48E-002
Cm245	2.12E+000	2.12E+000	-5.02E-003
Cm246	4.03E+000	4.26E+000	2.32E-001
Cm247	3.55E-001	3.80E-001	2.45E-002
Cm248	3.83E-001	4.34E-001	5.16E-002
Pu	1.27E+002	1.05E+002	-2.21E+001
Np	5.14E+000	3.72E+000	-1.43E+000
Am	1.72E+001	1.47E+001	-2.58E+000
Cm	1.94E+001	1.96E+001	2.88E-001
Total TRU	1.68E+002	1.43E+002	-2.58E+001

Charge, discharge, and production rate for a TRU+Cm 15th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.64E+001	1.56E+001	-8.16E-001
Pu239	3.77E+001	2.46E+001	-1.31E+001
Pu240	3.21E+001	2.61E+001	-5.97E+000
Pu241	1.21E+001	1.07E+001	-1.44E+000
Pu242	3.00E+001	2.87E+001	-1.29E+000
Np237	5.24E+000	3.77E+000	-1.47E+000
Am241	7.38E+000	5.06E+000	-2.31E+000
Am242m	4.41E-002	4.20E-002	-2.08E-003
Am243	9.86E+000	9.59E+000	-2.77E-001
Cm244	1.20E+001	1.21E+001	1.03E-001
Cm245	2.08E+000	2.09E+000	1.00E-002
Cm246	2.86E+000	3.19E+000	3.32E-001
Cm247	2.34E-001	2.68E-001	3.41E-002
Cm248	1.80E-001	2.29E-001	4.84E-002
Pu	1.28E+002	1.06E+002	-2.26E+001
Np	5.24E+000	3.77E+000	-1.47E+000
Am	1.73E+001	1.47E+001	-2.59E+000
Cm	1.74E+001	1.79E+001	5.27E-001
Total TRU	1.68E+002	1.42E+002	-2.62E+001

APPENDIX B

TRANSURANIC DESTRUCTION: TRU WITHOUT CURIUM

The amount of transuranics charged and discharged as well as the production rate for each of the 15 recycles for the TRU-Cm recycling strategy. The discharged fuel in the following tables is the amount of transuranics remaining after irradiation and the five year cooling period. All of the values have been normalized to the amount of electricity generated by the fuel assembly.

Charge, discharge, and production rate for a TRU-Cm 1st recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	3.42E+000	1.30E+001	9.60E+000
Pu239	8.27E+001	3.29E+001	-4.98E+001
Pu240	3.76E+001	3.29E+001	-4.70E+000
Pu241	8.70E+000	1.33E+001	4.65E+000
Pu242	9.98E+000	1.37E+001	3.67E+000
Np237	9.23E+000	5.34E+000	-3.89E+000
Am241	1.45E+001	7.36E+000	-7.16E+000
Am242m	1.01E-002	7.57E-002	6.57E-002
Am243	2.26E+000	4.44E+000	2.19E+000
Cm244	0.00E+000	2.66E+000	2.66E+000
Cm245	0.00E+000	3.69E-001	3.69E-001
Cm246	0.00E+000	4.41E-002	4.41E-002
Cm247	0.00E+000	8.27E-004	8.27E-004
Cm248	0.00E+000	5.82E-005	5.82E-005
Pu	1.42E+002	1.06E+002	-3.65E+001
Np	9.23E+000	5.34E+000	-3.89E+000
Am	1.68E+001	1.19E+001	-4.91E+000
Cm	0.00E+000	3.07E+000	3.07E+000
Total TRU	1.68E+002	1.26E+002	-4.23E+001

Charge, discharge, and production rate for a TRU-Cm 2nd recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.39E+001	1.76E+001	3.70E+000
Pu239	5.52E+001	3.00E+001	-2.52E+001
Pu240	4.30E+001	3.17E+001	-1.13E+001
Pu241	1.57E+001	1.37E+001	-2.00E+000
Pu242	1.63E+001	1.92E+001	2.84E+000
Np237	7.83E+000	5.03E+000	-2.81E+000
Am241	1.13E+001	7.22E+000	-4.05E+000
Am242m	7.84E-002	7.26E-002	-5.83E-003
Am243	5.05E+000	6.39E+000	1.34E+000
Cm244	0.00E+000	4.08E+000	4.08E+000
Cm245	0.00E+000	5.87E-001	5.87E-001
Cm246	0.00E+000	7.00E-002	7.00E-002
Cm247	0.00E+000	1.34E-003	1.34E-003
Cm248	0.00E+000	9.39E-005	9.39E-005
Pu	1.44E+002	1.12E+002	-3.19E+001
Np	7.83E+000	5.03E+000	-2.81E+000
Am	1.64E+001	1.37E+001	-2.71E+000
Cm	0.00E+000	4.74E+000	4.74E+000
Total TRU	1.68E+002	1.36E+002	-3.27E+001

Charge, discharge, and production rate for a TRU-Cm 3rd recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.84E+001	1.96E+001	1.18E+000
Pu239	4.85E+001	2.93E+001	-1.92E+001
Pu240	4.00E+001	2.96E+001	-1.05E+001
Pu241	1.56E+001	1.30E+001	-2.64E+000
Pu242	2.14E+001	2.28E+001	1.41E+000
Np237	7.08E+000	4.78E+000	-2.31E+000
Am241	1.04E+001	6.82E+000	-3.63E+000
Am242m	7.47E-002	6.85E-002	-6.26E-003
Am243	6.88E+000	7.64E+000	7.56E-001
Cm244	0.00E+000	4.99E+000	4.99E+000
Cm245	0.00E+000	7.15E-001	7.15E-001
Cm246	0.00E+000	8.60E-002	8.60E-002
Cm247	0.00E+000	1.65E-003	1.65E-003
Cm248	0.00E+000	1.17E-004	1.17E-004
Pu	1.44E+002	1.14E+002	-2.97E+001
Np	7.08E+000	4.78E+000	-2.31E+000
Am	1.74E+001	1.45E+001	-2.88E+000
Cm	0.00E+000	5.79E+000	5.79E+000
Total TRU	1.68E+002	1.39E+002	-2.91E+001

Charge, discharge, and production rate for a TRU-Cm 4th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.03E+001	2.01E+001	-1.15E-001
Pu239	4.64E+001	2.89E+001	-1.75E+001
Pu240	3.73E+001	2.82E+001	-9.14E+000
Pu241	1.48E+001	1.24E+001	-2.37E+000
Pu242	2.49E+001	2.52E+001	3.45E-001
Np237	6.69E+000	4.63E+000	-2.06E+000
Am241	9.84E+000	6.47E+000	-3.37E+000
Am242m	7.05E-002	6.42E-002	-6.34E-003
Am243	8.10E+000	8.46E+000	3.53E-001
Cm244	0.00E+000	5.58E+000	5.58E+000
Cm245	0.00E+000	7.93E-001	7.93E-001
Cm246	0.00E+000	9.65E-002	9.65E-002
Cm247	0.00E+000	1.86E-003	1.86E-003
Cm248	0.00E+000	1.33E-004	1.33E-004
Pu	1.44E+002	1.15E+002	-2.88E+001
Np	6.69E+000	4.63E+000	-2.06E+000
Am	1.80E+001	1.50E+001	-3.02E+000
Cm	0.00E+000	6.47E+000	6.47E+000
Total TRU	1.68E+002	1.41E+002	-2.74E+001

Charge, discharge, and production rate for a TRU-Cm 5th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.08E+001	2.01E+001	-6.88E-001
Pu239	4.56E+001	2.86E+001	-1.69E+001
Pu240	3.58E+001	2.75E+001	-8.31E+000
Pu241	1.41E+001	1.21E+001	-2.09E+000
Pu242	2.72E+001	2.69E+001	-3.43E-001
Np237	6.49E+000	4.55E+000	-1.94E+000
Am241	9.40E+000	6.24E+000	-3.16E+000
Am242m	6.61E-002	6.11E-002	-5.02E-003
Am243	8.90E+000	9.00E+000	9.22E-002
Cm244	0.00E+000	5.96E+000	5.96E+000
Cm245	0.00E+000	8.42E-001	8.42E-001
Cm246	0.00E+000	1.03E-001	1.03E-001
Cm247	0.00E+000	2.00E-003	2.00E-003
Cm248	0.00E+000	1.43E-004	1.43E-004
Pu	1.44E+002	1.15E+002	-2.84E+001
Np	6.49E+000	4.55E+000	-1.94E+000
Am	1.84E+001	1.53E+001	-3.07E+000
Cm	0.00E+000	6.91E+000	6.91E+000
Total TRU	1.68E+002	1.42E+002	-2.65E+001

Charge, discharge, and production rate for a TRU-Cm 6th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.08E+001	1.99E+001	-8.80E-001
Pu239	4.50E+001	2.84E+001	-1.66E+001
Pu240	3.49E+001	2.71E+001	-7.86E+000
Pu241	1.38E+001	1.19E+001	-1.92E+000
Pu242	2.88E+001	2.80E+001	-7.83E-001
Np237	6.38E+000	4.51E+000	-1.87E+000
Am241	9.12E+000	6.10E+000	-3.02E+000
Am242m	6.31E-002	5.92E-002	-3.82E-003
Am243	9.44E+000	9.37E+000	-7.21E-002
Cm244	0.00E+000	6.21E+000	6.21E+000
Cm245	0.00E+000	8.73E-001	8.73E-001
Cm246	0.00E+000	1.08E-001	1.08E-001
Cm247	0.00E+000	2.09E-003	2.09E-003
Cm248	0.00E+000	1.50E-004	1.50E-004
Pu	1.43E+002	1.15E+002	-2.81E+001
Np	6.38E+000	4.51E+000	-1.87E+000
Am	1.86E+001	1.55E+001	-3.10E+000
Cm	0.00E+000	7.20E+000	7.20E+000
Total TRU	1.68E+002	1.43E+002	-2.59E+001

Charge, discharge, and production rate for a TRU-Cm 7th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.06E+001	1.97E+001	-9.08E-001
Pu239	4.47E+001	2.82E+001	-1.64E+001
Pu240	3.44E+001	2.68E+001	-7.62E+000
Pu241	1.36E+001	1.17E+001	-1.83E+000
Pu242	3.00E+001	2.89E+001	-1.07E+000
Np237	6.32E+000	4.49E+000	-1.84E+000
Am241	8.96E+000	6.02E+000	-2.94E+000
Am242m	6.12E-002	5.81E-002	-3.09E-003
Am243	9.80E+000	9.63E+000	-1.77E-001
Cm244	0.00E+000	6.39E+000	6.39E+000
Cm245	0.00E+000	8.95E-001	8.95E-001
Cm246	0.00E+000	1.11E-001	1.11E-001
Cm247	0.00E+000	2.15E-003	2.15E-003
Cm248	0.00E+000	1.55E-004	1.55E-004
Pu	1.43E+002	1.15E+002	-2.79E+001
Np	6.32E+000	4.49E+000	-1.84E+000
Am	1.88E+001	1.57E+001	-3.12E+000
Cm	0.00E+000	7.39E+000	7.39E+000
Total TRU	1.68E+002	1.43E+002	-2.54E+001

Charge, discharge, and production rate for a TRU-Cm 8th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.03E+001	1.94E+001	-8.71E-001
Pu239	4.44E+001	2.81E+001	-1.63E+001
Pu240	3.41E+001	2.66E+001	-7.48E+000
Pu241	1.34E+001	1.17E+001	-1.77E+000
Pu242	3.08E+001	2.96E+001	-1.28E+000
Np237	6.29E+000	4.48E+000	-1.81E+000
Am241	8.86E+000	5.97E+000	-2.89E+000
Am242m	6.00E-002	5.74E-002	-2.62E-003
Am243	1.01E+001	9.81E+000	-2.45E-001
Cm244	0.00E+000	6.51E+000	6.51E+000
Cm245	0.00E+000	9.09E-001	9.09E-001
Cm246	0.00E+000	1.13E-001	1.13E-001
Cm247	0.00E+000	2.19E-003	2.19E-003
Cm248	0.00E+000	1.58E-004	1.58E-004
Pu	1.43E+002	1.15E+002	-2.77E+001
Np	6.29E+000	4.48E+000	-1.81E+000
Am	1.90E+001	1.58E+001	-3.14E+000
Cm	0.00E+000	7.53E+000	7.53E+000
Total TRU	1.68E+002	1.43E+002	-2.51E+001

Charge, discharge, and production rate for a TRU-Cm 9th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	2.01E+001	1.93E+001	-8.23E-001
Pu239	4.42E+001	2.80E+001	-1.62E+001
Pu240	3.39E+001	2.65E+001	-7.40E+000
Pu241	1.33E+001	1.16E+001	-1.74E+000
Pu242	3.15E+001	3.00E+001	-1.42E+000
Np237	6.27E+000	4.47E+000	-1.80E+000
Am241	8.79E+000	5.93E+000	-2.86E+000
Am242m	5.92E-002	5.69E-002	-2.38E-003
Am243	1.02E+001	9.95E+000	-2.91E-001
Cm244	0.00E+000	6.60E+000	6.60E+000
Cm245	0.00E+000	9.20E-001	9.20E-001
Cm246	0.00E+000	1.15E-001	1.15E-001
Cm247	0.00E+000	2.22E-003	2.22E-003
Cm248	0.00E+000	1.60E-004	1.60E-004
Pu	1.43E+002	1.15E+002	-2.76E+001
Np	6.27E+000	4.47E+000	-1.80E+000
Am	1.91E+001	1.59E+001	-3.15E+000
Cm	0.00E+000	7.63E+000	7.63E+000
Total TRU	1.68E+002	1.44E+002	-2.49E+001

Charge, discharge, and production rate for a TRU-Cm 10^{th} recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.99E+001	1.91E+001	-7.81E-001
Pu239	4.40E+001	2.79E+001	-1.61E+001
Pu240	3.38E+001	2.65E+001	-7.34E+000
Pu241	1.33E+001	1.16E+001	-1.72E+000
Pu242	3.19E+001	3.04E+001	-1.53E+000
Np237	6.26E+000	4.47E+000	-1.79E+000
Am241	8.74E+000	5.90E+000	-2.84E+000
Am242m	5.87E-002	5.65E-002	-2.24E-003
Am243	1.04E+001	1.01E+001	-3.24E-001
Cm244	0.00E+000	6.66E+000	6.66E+000
Cm245	0.00E+000	9.27E-001	9.27E-001
Cm246	0.00E+000	1.16E-001	1.16E-001
Cm247	0.00E+000	2.24E-003	2.24E-003
Cm248	0.00E+000	1.62E-004	1.62E-004
Pu	1.43E+002	1.16E+002	-2.74E+001
Np	6.26E+000	4.47E+000	-1.79E+000
Am	1.92E+001	1.60E+001	-3.16E+000
Cm	0.00E+000	7.71E+000	7.71E+000
Total TRU	1.68E+002	1.44E+002	-2.47E+001

Charge, discharge, and production rate for a TRU-Cm 11th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.98E+001	1.90E+001	-7.51E-001
Pu239	4.39E+001	2.79E+001	-1.60E+001
Pu240	3.37E+001	2.64E+001	-7.30E+000
Pu241	1.32E+001	1.15E+001	-1.70E+000
Pu242	3.23E+001	3.07E+001	-1.61E+000
Np237	6.25E+000	4.47E+000	-1.78E+000
Am241	8.70E+000	5.88E+000	-2.82E+000
Am242m	5.84E-002	5.62E-002	-2.17E-003
Am243	1.05E+001	1.01E+001	-3.49E-001
Cm244	0.00E+000	6.71E+000	6.71E+000
Cm245	0.00E+000	9.33E-001	9.33E-001
Cm246	0.00E+000	1.17E-001	1.17E-001
Cm247	0.00E+000	2.26E-003	2.26E-003
Cm248	0.00E+000	1.63E-004	1.63E-004
Pu	1.43E+002	1.16E+002	-2.74E+001
Np	6.25E+000	4.47E+000	-1.78E+000
Am	1.92E+001	1.61E+001	-3.17E+000
Cm	0.00E+000	7.76E+000	7.76E+000
Total TRU	1.68E+002	1.44E+002	-2.46E+001

Charge, discharge, and production rate for a TRU-Cm 12th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.96E+001	1.89E+001	-7.20E-001
Pu239	4.38E+001	2.78E+001	-1.60E+001
Pu240	3.36E+001	2.64E+001	-7.26E+000
Pu241	1.32E+001	1.15E+001	-1.69E+000
Pu242	3.26E+001	3.09E+001	-1.68E+000
Np237	6.25E+000	4.47E+000	-1.78E+000
Am241	8.68E+000	5.87E+000	-2.81E+000
Am242m	5.81E-002	5.61E-002	-2.07E-003
Am243	1.06E+001	1.02E+001	-3.66E-001
Cm244	0.00E+000	6.75E+000	6.75E+000
Cm245	0.00E+000	9.38E-001	9.38E-001
Cm246	0.00E+000	1.17E-001	1.17E-001
Cm247	0.00E+000	2.27E-003	2.27E-003
Cm248	0.00E+000	1.64E-004	1.64E-004
Pu	1.43E+002	1.16E+002	-2.73E+001
Np	6.25E+000	4.47E+000	-1.78E+000
Am	1.93E+001	1.61E+001	-3.18E+000
Cm	0.00E+000	7.80E+000	7.80E+000
Total TRU	1.68E+002	1.44E+002	-2.45E+001

Charge, discharge, and production rate for a TRU-Cm 13th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.96E+001	1.89E+001	-7.00E-001
Pu239	4.37E+001	2.78E+001	-1.59E+001
Pu240	3.35E+001	2.63E+001	-7.24E+000
Pu241	1.32E+001	1.15E+001	-1.68E+000
Pu242	3.28E+001	3.11E+001	-1.72E+000
Np237	6.24E+000	4.47E+000	-1.77E+000
Am241	8.66E+000	5.86E+000	-2.80E+000
Am242m	5.79E-002	5.59E-002	-2.02E-003
Am243	1.06E+001	1.02E+001	-3.79E-001
Cm244	0.00E+000	6.78E+000	6.78E+000
Cm245	0.00E+000	9.41E-001	9.41E-001
Cm246	0.00E+000	1.18E-001	1.18E-001
Cm247	0.00E+000	2.28E-003	2.28E-003
Cm248	0.00E+000	1.65E-004	1.65E-004
Pu	1.43E+002	1.16E+002	-2.73E+001
Np	6.24E+000	4.47E+000	-1.77E+000
Am	1.93E+001	1.61E+001	-3.18E+000
Cm	0.00E+000	7.84E+000	7.84E+000
Total TRU	1.68E+002	1.44E+002	-2.44E+001

Charge, discharge, and production rate for a TRU-Cm 14th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.95E+001	1.88E+001	-6.84E-001
Pu239	4.36E+001	2.78E+001	-1.59E+001
Pu240	3.35E+001	2.63E+001	-7.22E+000
Pu241	1.31E+001	1.15E+001	-1.67E+000
Pu242	3.30E+001	3.12E+001	-1.76E+000
Np237	6.24E+000	4.47E+000	-1.77E+000
Am241	8.64E+000	5.85E+000	-2.79E+000
Am242m	5.78E-002	5.58E-002	-1.98E-003
Am243	1.07E+001	1.03E+001	-3.89E-001
Cm244	0.00E+000	6.80E+000	6.80E+000
Cm245	0.00E+000	9.43E-001	9.43E-001
Cm246	0.00E+000	1.18E-001	1.18E-001
Cm247	0.00E+000	2.29E-003	2.29E-003
Cm248	0.00E+000	1.65E-004	1.65E-004
Pu	1.43E+002	1.16E+002	-2.72E+001
Np	6.24E+000	4.47E+000	-1.77E+000
Am	1.94E+001	1.62E+001	-3.18E+000
Cm	0.00E+000	7.86E+000	7.86E+000
Total TRU	1.68E+002	1.44E+002	-2.43E+001

Charge, discharge, and production rate for a TRU-Cm 15th recycle.

Isotope	Charge (kg/TWhe)	Discharge (kg/TWhe)	Net Production (kg/TWhe)
Pu238	1.95E+001	1.88E+001	-6.73E-001
Pu239	4.36E+001	2.77E+001	-1.59E+001
Pu240	3.35E+001	2.63E+001	-7.21E+000
Pu241	1.31E+001	1.14E+001	-1.67E+000
Pu242	3.31E+001	3.13E+001	-1.79E+000
Np237	6.24E+000	4.47E+000	-1.77E+000
Am241	8.63E+000	5.84E+000	-2.79E+000
Am242m	5.77E-002	5.57E-002	-1.97E-003
Am243	1.07E+001	1.03E+001	-3.97E-001
Cm244	0.00E+000	6.81E+000	6.81E+000
Cm245	0.00E+000	9.45E-001	9.45E-001
Cm246	0.00E+000	1.18E-001	1.18E-001
Cm247	0.00E+000	2.29E-003	2.29E-003
Cm248	0.00E+000	1.66E-004	1.66E-004
Pu	1.43E+002	1.16E+002	-2.72E+001
Np	6.24E+000	4.47E+000	-1.77E+000
Am	1.94E+001	1.62E+001	-3.19E+000
Cm	0.00E+000	7.88E+000	7.88E+000
Total TRU	1.68E+002	1.44E+002	-2.43E+001

APPENDIX C

TRANSURANICS CHARGE AT THE BEGINNING OF EACH CYCLE

The amount (in kg) of transuranics included initially at the beginning of each recycle is included in the tables below.

Amount of Neptunium included initially for the TRU+Cm fuel.

Cycle	Np237	Np238	Np239
1	1.80E+000	3.57E-010	3.80E-007
2	1.50E+000	2.76E-009	8.43E-007
3	1.31E+000	2.50E-009	1.14E-006
4	1.20E+000	2.23E-009	1.34E-006
5	1.13E+000	1.99E-009	1.46E-006
6	1.09E+000	1.82E-009	1.54E-006
7	1.06E+000	1.72E-009	1.59E-006
8	1.05E+000	1.65E-009	1.62E-006
9	1.04E+000	1.61E-009	1.65E-006
10	1.03E+000	1.57E-009	1.66E-006
11	1.02E+000	1.55E-009	1.68E-006
12	1.01E+000	1.53E-009	1.68E-006
13	1.01E+000	1.51E-009	1.69E-006
14	1.01E+000	1.49E-009	1.69E-006
15	1.01E+000	1.48E-009	1.69E-006

Amount of Neptunium included initially for the TRU-Cm fuel.

Cycle	Np237	Np238	Np239
1	1.81E+000	3.58E-010	3.81E-007
2	1.53E+000	2.79E-009	8.52E-007
3	1.39E+000	2.66E-009	1.16E-006
4	1.31E+000	2.51E-009	1.37E-006
5	1.27E+000	2.36E-009	1.50E-006
6	1.25E+000	2.25E-009	1.59E-006
7	1.24E+000	2.18E-009	1.65E-006
8	1.23E+000	2.14E-009	1.70E-006
9	1.23E+000	2.11E-009	1.73E-006
10	1.23E+000	2.09E-009	1.75E-006
11	1.23E+000	2.08E-009	1.77E-006
12	1.22E+000	2.07E-009	1.78E-006
13	1.22E+000	2.06E-009	1.79E-006
14	1.22E+000	2.06E-009	1.80E-006
15	1.22E+000	2.06E-009	1.80E-006

Amount of Plutonium included initially for the TRU+Cm fuel.

Cycle	Pu238	Pu239	Pu240	Pu241	Pu242
1	6.69E-001	1.61E+001	7.34E+000	1.70E+000	1.95E+000
2	2.71E+000	1.05E+001	8.28E+000	3.04E+000	3.16E+000
3	3.52E+000	8.91E+000	7.59E+000	2.97E+000	4.11E+000
4	3.78E+000	8.33E+000	7.01E+000	2.76E+000	4.74E+000
5	3.76E+000	8.03E+000	6.69E+000	2.60E+000	5.14E+000
6	3.65E+000	7.82E+000	6.52E+000	2.51E+000	5.41E+000
7	3.52E+000	7.66E+000	6.43E+000	2.46E+000	5.60E+000
8	3.40E+000	7.55E+000	6.37E+000	2.42E+000	5.73E+000
9	3.30E+000	7.46E+000	6.33E+000	2.40E+000	5.82E+000
10	3.22E+000	7.39E+000	6.30E+000	2.38E+000	5.88E+000
11	3.17E+000	7.33E+000	6.27E+000	2.36E+000	5.93E+000
12	3.12E+000	7.29E+000	6.24E+000	2.35E+000	5.96E+000
13	3.09E+000	7.26E+000	6.22E+000	2.33E+000	5.99E+000
14	3.06E+000	7.22E+000	6.20E+000	2.33E+000	6.00E+000
15	3.04E+000	7.21E+000	6.18E+000	2.32E+000	6.00E+000

Amount of Plutonium included initially for the TRU-Cm fuel.

Cycle	Pu238	Pu239	Pu240	Pu241	Pu242
1	6.71E-001	1.62E+001	7.36E+000	1.70E+000	1.96E+000
2	2.73E+000	1.08E+001	8.42E+000	3.07E+000	3.20E+000
3	3.60E+000	9.49E+000	7.84E+000	3.06E+000	4.19E+000
4	3.97E+000	9.10E+000	7.32E+000	2.89E+000	4.87E+000
5	4.08E+000	8.93E+000	7.01E+000	2.77E+000	5.33E+000
6	4.07E+000	8.83E+000	6.84E+000	2.70E+000	5.65E+000
7	4.03E+000	8.75E+000	6.75E+000	2.66E+000	5.87E+000
8	3.98E+000	8.70E+000	6.69E+000	2.63E+000	6.04E+000
9	3.93E+000	8.66E+000	6.65E+000	2.61E+000	6.16E+000
10	3.90E+000	8.62E+000	6.62E+000	2.60E+000	6.26E+000
11	3.87E+000	8.59E+000	6.60E+000	2.59E+000	6.33E+000
12	3.85E+000	8.58E+000	6.59E+000	2.58E+000	6.39E+000
13	3.83E+000	8.56E+000	6.57E+000	2.58E+000	6.43E+000
14	3.82E+000	8.55E+000	6.57E+000	2.57E+000	6.46E+000
15	3.81E+000	8.54E+000	6.56E+000	2.57E+000	6.49E+000

Amount of Americium included initially for the TRU+Cm fuel.

Cycle	Am241	Am242m	Am243
1	2.84E+000	1.96E-003	4.41E-001
2	2.15E+000	1.52E-002	9.79E-001
3	1.93E+000	1.38E-002	1.33E+000
4	1.75E+000	1.23E-002	1.55E+000
5	1.64E+000	1.09E-002	1.70E+000
6	1.56E+000	1.00E-002	1.79E+000
7	1.51E+000	9.46E-003	1.85E+000
8	1.48E+000	9.08E-003	1.89E+000
9	1.46E+000	8.83E-003	1.91E+000
10	1.45E+000	8.65E-003	1.93E+000
11	1.43E+000	8.50E-003	1.95E+000
12	1.42E+000	8.39E-003	1.96E+000
13	1.41E+000	8.28E-003	1.96E+000
14	1.41E+000	8.21E-003	1.97E+000
15	1.40E+000	8.15E-003	1.97E+000

Amount of Americium included initially for the TRU-Cm fuel.

Cycle	Am241	Am242m	Am243
1	2.84E+000	1.97E-003	4.42E-001
2	2.21E+000	1.54E-002	9.89E-001
3	2.05E+000	1.47E-002	1.35E+000
4	1.93E+000	1.38E-002	1.59E+000
5	1.84E+000	1.30E-002	1.75E+000
6	1.79E+000	1.24E-002	1.85E+000
7	1.75E+000	1.20E-002	1.92E+000
8	1.74E+000	1.18E-002	1.97E+000
9	1.72E+000	1.16E-002	2.01E+000
10	1.71E+000	1.15E-002	2.03E+000
11	1.71E+000	1.15E-002	2.05E+000
12	1.70E+000	1.14E-002	2.07E+000
13	1.70E+000	1.14E-002	2.08E+000
14	1.69E+000	1.13E-002	2.09E+000
15	1.69E+000	1.13E-002	2.09E+000

Amount of Curium included initially for the TRU+Cm fuel.

Cycle	Cm242	Cm243	Cm244	Cm245
1	5.13E-006	9.80E-004	8.23E-002	1.13E-002
2	1.50E-004	1.44E-002	5.75E-001	8.51E-002
3	1.31E-004	1.39E-002	1.06E+000	1.86E-001
4	1.18E-004	1.29E-002	1.46E+000	2.66E-001
5	1.08E-004	1.19E-002	1.76E+000	3.20E-001
6	1.01E-004	1.12E-002	1.98E+000	3.55E-001
7	9.62E-005	1.07E-002	2.13E+000	3.78E-001
8	9.34E-005	1.04E-002	2.23E+000	3.92E-001
9	9.15E-005	1.02E-002	2.30E+000	4.01E-001
10	9.02E-005	1.01E-002	2.35E+000	4.07E-001
11	8.92E-005	1.00E-002	2.39E+000	4.11E-001
12	8.83E-005	9.95E-003	2.41E+000	4.14E-001
13	8.76E-005	9.88E-003	2.43E+000	4.15E-001
14	8.71E-005	9.83E-003	2.44E+000	4.16E-001
15	8.66E-005	9.78E-003	2.45E+000	4.16E-001

Amount of Curium included initially for the TRU+Cm fuel.

Cycle	Cm246	Cm247	Cm248
1	1.26E-003	1.55E-005	1.04E-006
2	1.35E-002	3.52E-004	4.07E-005
3	4.96E-002	1.91E-003	3.36E-004
4	1.08E-001	5.32E-003	1.33E-003
5	1.79E-001	1.05E-002	3.49E-003
6	2.58E-001	1.69E-002	7.11E-003
7	3.38E-001	2.40E-002	1.23E-002
8	4.16E-001	3.14E-002	1.88E-002
9	4.91E-001	3.87E-002	2.66E-002
10	5.61E-001	4.58E-002	3.54E-002
11	6.25E-001	5.25E-002	4.48E-002
12	6.85E-001	5.87E-002	5.47E-002
13	7.39E-001	6.43E-002	6.48E-002
14	7.89E-001	6.96E-002	7.50E-002
15	8.34E-001	7.43E-002	8.50E-002

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